Indian Point
3



# Improved

# Technical Specifications

# Conversion Submittal

Volume 8

New York Power
Authority

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.11:
## "Pressurizer Power Operated Relief Valves (PORVs)"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

# 3.4 REACTOR COOLANT SYSTEM (RCS)

## 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------------ NOTES ------------------------------------

1. Separate Condition entry is allowed for each PORV.
2. LCO 3.0.4 is not applicable.

-------------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more PORVs inoperable and capable of being manually cycled. | A.1 | Close and maintain power to associated block valve. | 1 hour |
| B. One PORV inoperable and not capable of being manually cycled. | B.1 | Close associated block valve. | 1 hour |
| | <u>AND</u> | | |
| | B.2 | Remove power from associated block valve. | 1 hour |
| | <u>AND</u> | | |
| | B.3 | Restore PORV to OPERABLE status. | 7 days |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| C. One block valve inoperable. | C.1 | Place associated PORV in manual control. | 1 hour |
| | <u>AND</u> | | |
| | C.2 | Restore block valve to OPERABLE status. | 7 days |
| D. Required Action and associated Completion Time of Condition A, B, or C not met. | D.1 | Be in MODE 3. | 6 hours |
| | <u>AND</u> | | |
| | D.2 | Be in MODE 4. | 12 hours |
| E. Two PORVs inoperable and not capable of being manually cycled. | E.1 | Close associated block valves. | 1 hour |
| | <u>AND</u> | | |
| | E.2 | Remove power from associated block valves. | 1 hour |
| | <u>AND</u> | | |
| | E.3 | Be in MODE 3. | 6 hours |
| | <u>AND</u> | | |
| | E.4 | Be in MODE 4. | 12 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| F. More than one block valve inoperable. | F.1 | Place associated PORVs in manual control. | 1 hour |
| | AND | | |
| | F.2 | Restore one block valve to OPERABLE status. | 2 hours |
| G. Required Action and associated Completion Time of Condition F not met. | G.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | G.2 | Be in MODE 4. | 12 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.11.1 | ------------------NOTE------------------<br>Not required to be met with block valve closed in accordance with the Required Action of Condition B or E.<br>----------------------------------------<br>Perform a complete cycle of each block valve. | 92 days |
| SR 3.4.11.2 | Perform a complete cycle of each PORV. | 24 months |

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are nitrogen operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal and alternate pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

Electrical power neeeded to support the PORVs, their block valves, and their controls is supplied from the vital buses that normally receive power from offsite power sources, but is also capable of being supplied from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

The plant has two PORVs, each having a design relief capacity of 179,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer

BASES

BACKGROUND (Continued)

Pressure - High reactor trip setpoint following a step reduction of 50% of full load with steam dump and automatic reactor control operation. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal and alternate pressurizer spray are not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs or auxiliary spray are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical (Ref. 2). By assuming PORV manual actuation, the DNBR calculation is more conservative although not required to meet safety limits. As such, this actuation is not required to mitigate these events, and PORV automatic operation is not an assumed safety function.

Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36.

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

BASES

LCO (continued)

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage. Satisfying the LCO helps minimize challenges to fission product barriers.

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4,

BASES

APPLICABILITY (continued)

5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits entry into MODES 1, 2, and 3. This exception to LCO requirements is normally used to perform cycling of the PORVs or block valves to verify their OPERABLE status because testing is not performed in lower MODES.

A.1

PORVs may be inoperable and capable of being manually cycled (e.g., excessive seat leakage). In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valve is required to be closed, but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. This permits operation of the plant until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, and B.3

If one PORV is inoperable and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Time of 1 hour is reasonable, based on challenges to the PORVs during this time period, and provide

ACTIONS

B.1, B.2, and B.3 (continued)

the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 7 days is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in the closed position (i.e., switch in manual control). The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 7 days to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs may not be capable of mitigating an overpressure event if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 7 days, the power will be restored and the PORV restored. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at

ACTIONS

D.1 and D.2 (continued)

least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

E.1, E.2, E.3 and E.4

If more than one PORV is inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

F.1 and F.2

If more than one block valve is inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control (i.e., closed position) and restore at least one block valve within 2 hours. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least

BASES

ACTIONS

G.1 and G.2 (continued)

MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME Code, Section XI (Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is important because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an inoperable PORV that is not capable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is 7 days, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status.

The Note modifies this SR by stating that it is not required to be met with the block valve closed, in accordance with the Required Action of this LCO.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice.

BASES

REFERENCES

1. Regulatory Guide 1.32, February 1977.
2. FSAR, Section 14.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.11:
## "Pressurizer Power Operated Relief Valves (PORVs)"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.1-4 | 170 | 170 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-8 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| T 4.1-3(2) | 148 | 148 | No TSCRs | No TSCRs for this Page | N/A |

Add LCO 3.4.11, Actions Note 1 — A.5

Add LCO 3.4.11, Actions Note 2 — A.4

SEE ITS 3.4.10

2. Safety Valves

a. At least one pressurizer code safety valve shall be operable, or an opening greater than or equal to the size of one code safety valve flange to allow for pressure relief, whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is above the cold shutdown condition except during reactor coolant system hydrostatic tests and/or safety valve settings.

c. The pressurizer code safety valve lift setting shall be set at 2485 psig with ±1% allowance for error.

SEE ITS 3.4.9

3. Pressurizer Heaters

Whenever the reactor is above the hot shutdown condition, the pressurizer shall be operable with at least 150 kw of pressurizer heaters.

a. With less than 150 kw of pressurizer heaters operable, restore the required inoperable heaters within 72 hours or be in at least hot shutdown within an additional 6 hours.

LCO 3.4.11 ~~4.~~ Power Operated Relief Valves

LCO 3.4.11 Req. Act A.1

Mode 1, 2, 3 — M.2

~~Whenever the reactor coolant system is above 400°F~~, the power operated relief valves (PORVs) shall be operable or their associated block valves closed. within one hour — A.3

a. If the block valve is closed because of an inoperable PORV, the control power for the block valve must be removed.

b. If the above conditions cannot be satisfied within 1 hour, be in at least hot shutdown within 6 hours and in cold shutdown within the following 30 hours.

LCO 3.4.11 ~~5.~~ Power Operated Relief Block Valves

Mode 1, 2, 3 — M.2

~~Whenever the reactor coolant system is above 400°F~~, the motor operated block valves shall be operable or closed.

a. If the block valve is inoperable, the control power is to be removed.

b. If the above conditions cannot be satisfied within 1 hour be in at least hot shutdown within the following 30 hours.

~~6. Deleted~~

Add Conditions C, D, F and G and assoc. Req. Act — M.1

Add Conditions B, D, E and G and assoc. Req. Act — M.1

A.1

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal off possible RCS leakage paths.

Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor vessel head vent path ensures that capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

The OPS is designed to relieve the RCS pressure for certain unlikely incidents to prevent the peak RCS pressure from exceeding the limits established in Reg. Guide 1.99, Revision 2. The OPS is considered to be operable when the minimum number of required channels (per Table 3.5-3) are available to open the PORVs upon receipt of a high pressure signal which is based upon RCS $T_{cold}$, as shown in Figure 3.1.A-2. (The happy face icon contained on this and other Technical Specification figures indicates the side of the applicable curve in which operation is permissible. Conversely, the sad face icon indicates the side of the applicable curve in which operation is prohibited.) The OPS setpoint is based upon a comparative analysis of References 5 and 9, with allowances for metal/fluid temperature differences (as described below) and for the static head due to elevation differences and dynamic head effect of the operation of the reactor coolant and RHR pumps. "Arming" means that the motor operated block valve (MOV) is in the open position. This can be accomplished either automatically by the OPS when the RCS temperature is less than or equal to 319°F or manually by the control room operator.

TABLE 4.1-3 (Sheet 2 of 2)

| | | | |
|---|---|---|---|
| SEE ITS 3.4.14 | 13. RHR Valves 730 and 731 | Automatic isolation and interlock action | 24M |
| SR 3.4.11.1 | 14. PORV Block Valves | Operability through 1 complete cycle of full travel | Quarterly (see Note 1) 92 days |
| SR 3.4.11.2 | 15. PORV Valves | Operability | 24M |
| SEE RELOCATED | 16. Reactor Vessel Head Vents | Operability | 24M |

24M - At least once per 24 months

Note 1. ~~If the block valve is shut due to a leaking or inoperable PORV, Block Valve operability will be checked the next time the plant is in cold shutdown.~~ A.6

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.11:
## "Pressurizer Power Operated Relief Valves (PORVs)"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.4.11 - Pressurizer Power Operated Relief Valves (PORVs)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases that are designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRS) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.1.A.4 and CTS 3.1.A.4.a identify requirements for inoperable PORVs without identifying any completion time; therefore, the time to initiate action is assumed to be immediately in accordance with CTS 3.0. ITS LCO 3.4.11, Conditions A and B, establish the similar requirements for inoperable PORVs (See ITS 3.4.11, DOC M.1) but specify a 1 hour completion time. This is an administrative change with no impact on safety because the ITS completion time is a reasonable interpretation of the existing requirements based on plant operating experience that closure is accomplished within approximately 1 hour.

A.4 ITS LCO 3.4.11, Note 2, is added to provide the allowance that ITS LCO 3.0.4 is not applicable to PORVs and block valves. This allowance permits entry into Modes 1, 2, and 3 if one or both PORVs are inoperable (pressure relieving function lost) but capable of being manually cycled (manual venting function maintained). Additionally, this allowance permits entry into Modes 1, 2, and 3 prior to performing the required cycling of the PORVs or block valves to verify their Operability status.

This note is needed because it allows entry into the Modes where ITS LCO 3.4.11 is applicable while implementing Required Actions for one or both PORVs are inoperable (pressure relieving function lost) but capable of being manually cycled (manual venting function maintained). This change is acceptable because operation may continue in this condition for an unlimited period of time. This is an administrative change with no impact on safety because there is no equivalent to LCO 3.0.4 in the CTS; therefore, providing an exception results in no changes to the existing requirements. The justification for adding LCO 3.0.4 is addressed in Discussion of Changes for ITS Section 3.0.

A.5 ITS LCO 3.4.11, Note 1, is added to clarify that separate Condition entry is allowed for each PORV. This note is needed to clarify that both pressurizer PORVs and associated block valves are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). This note provides more explicit instructions for proper application of the Actions for ITS compliance. In conjunction with the ITS Specification 1.3, "Completion Times," this Note provides direction consistent with the intent of the ITS Actions for inoperable PORVs or block valves. This Note ensures that a specified period of time to verify or restore compliance with requirements for each inoperable PORV or block valve. This is an administrative change with no impact on safety because it is consistent with changes to the Conditions and associated Required Actions for PORVs and block valves being added by ITS (See ITS 3.4.11, DOC M.1).

A.6 CTS Table 4.1-1, Item 14, Note 1, specifies that the requirement to cycle the block valve every quarter may be deferred until the next time the plant is in cold shutdown if the block valve is closed because of a leaking or inoperable PORV. ITS SR 3.4.11.1, Note, provides essentially the same allowance; however, the benefit afforded by this note is significantly reduced because of new requirements to restore the venting and isolation function of an inoperable PORV within 7 days (See ITS 3.4.11, DOC M.1) (versus CTS allowing unlimited operation with one or more PORVs with inoperable manual venting function). This is an administrative change with no impact on safety because the reduction in the allowance is caused by changes to required actions which are justified elsewhere (See ITS 3.4.11, DOC M.1).

MORE RESTRICTIVE

M.1 CTS 3.1.A.4, Power Operated Relief Valves (PORVs), and CTS 3.1.A.5, Power Operated Relief Block Valves, require that the PORVs and block valves are Operable. However, CTS 3.1.A.4 and CTS 3.1.A.5 establish Required Actions (consistent with the Safety Evaluation Report for Amendment 38) that assume the sole safety function of the PORVs and associated block valves (except when performing the LTOP function) is vent path isolation if a PORV fails open. Specifically, CTS 3.1.A.4 and CTS 3.1.A.5 allow unlimited operation with both PORVs and/or both block valves inoperable as long as the vent path is isolated (i.e., CTS allows a complete loss of manual venting function and a complete loss of the PORV pressure relief function for an unlimited period of time).

ITS LCO 3.4.11 maintains the requirement that PORVs and associated block valves must be Operable; however, LCO 3.4.11, Conditions and associated Required Actions are established to ensure that PORVs and associated block valves provide the following functions: a single failure proof method of manually venting the RCS via the pressurizer which may be needed to respond to a SGTR event; and, a single failure proof method of isolating that vent path which may be needed to respond to a stuck open PORV. Therefore, ITS 3.4.11 adds Conditions and Required Actions when a PORV and/or block valve is inoperable for venting, inoperable for isolation, or inoperable for both functions. The following Conditions and associated Required Actions are added:

a. Condition A and associated Required Actions maintain the allowance in CTS 3.1.A.4.a for unlimited operation with the pressure relief function of one or both PORVs inoperable; however, Required Action A.1 reverses CTS 3.1.A.4.a by requiring that power to a block valve must be maintained when the block valve is closed in response to a PORV that is inoperable (pressure relieving function lost) but capable of being manually cycled. This change is needed because maintaining power to the block valve ensures that the manual venting function is maintained because both the PORV and block valve remain capable of being manually cycled. This change is acceptable because the vent path is still protected by redundant isolation function and could tolerate a failure to close of either the PORV or the block valve during the manual venting operation.

b. Condition B and associated Required Actions are added to address one PORV inoperable and not capable of being manually cycled (loss of redundancy of the manual venting function). Required Action B.1 ensures that the inoperable PORV is promptly isolated. Required Action B.2, which removes power from the block valve closed by B.1, ensures that the block valve will not be opened when the redundant isolation function for that vent path (i.e., the PORV) is not Operable. Required Action B.3 requires restoration of redundant manual venting function within 7 days. This change is needed and is acceptable because of the following: it ensures that vent path isolation function is not challenged when redundant isolation function is not Operable; and, it ensures that redundant manual venting function is restored within 7 days.

c. Condition C and associated Required Actions are added to address one block valve inoperable (i.e., not capable of being manually cycled). Required Action C.1 places the PORV in manual control so that the PORV will not open on an overpressure condition because the block valve is not available to isolate the line if the PORV fails open. Required Action C.2 requires restoration of redundant manual isolation function within 7 days. This change is needed and is acceptable because of the following: it ensures that vent path isolation function is not challenged when redundant isolation function is not Operable; and, it ensures that redundant manual isolation function is restored within 7 days.

d. Condition D and associated Required Actions are added to require plant shutdown if required compensatory action for loss of redundant venting and/or isolation function are not implemented or if redundant venting and/or isolation function are not restored within 7 days.

e. Condition E and associated Required Actions are added to address both of the PORVs inoperable and not capable of being manually cycled (i.e., loss of venting function and/or loss of redundancy for the isolation function). Required Actions compensate for the loss of redundant isolation function by isolating the PORV within one hour and removing power from the closed block valve to ensure that the block valve will not be opened when the redundant isolation function for that vent path (i.e., the PORV) is not Operable. Additionally, plant shutdown is initiated because of the loss of manual venting function. This change is needed and acceptable because it ensures appropriate compensatory action for loss of redundancy for the isolation function and requires plant shutdown for loss of manual venting function.

f. Condition F and associated Required Actions are added to address both of the block valves inoperable (i.e., not capable of being manually cycled). Required Action F.1 places the PORV in manual control so that the PORVs will not open on an overpressure condition because the block valve is not available to isolate the line if the PORV fails open. Required Action F.2 requires restoration of redundant manual isolation function to at least one vent path within 2 hours. This change is needed and is acceptable because of the following: it ensures that vent path isolation function is not challenged when redundant isolation function is not Operable; and, it ensures that redundant manual isolation function is restored for at least one vent path within 2 hours.

g. Condition G and associated Required Actions are added to require plant shutdown if there is a loss of either the manual venting function or isolation function of the vent path.

These more restrictive changes are acceptable because they do not introduce any operation which is un-analyzed while establishing Conditions and associated Required Actions that ensure that PORVs and associated block valves provide both a single failure proof method of venting the RCS and a single failure proof method of isolating that vent path. Therefore, this change has no significant adverse impact on safety.

M.2 CTS 3.1.A.4 and CTS 3.1.A.5 require that the PORVs and associated block valves are Operable "whenever the reactor coolant system is above 400°F." ITS LCO 3.4.11, Pressurizer PORVs, require that the PORVs and associated block valves are Operable in Modes 1, 2 and 3 (i.e, whenever the reactor coolant system is above 350°F). This change is needed because PORV and block valve Operability is required for both a venting function and a vent path isolation function. PORVs and associated block valves must be operable in Mode 1, 2 and 3 both to ensure the ability to isolate a stuck open PORV and to minimize challenges to the pressurizer safety valves. This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring that PORVs and associated block valves are available to perform their venting function, pressure relief function and vent path isolation function over a wider range of plant conditions. Therefore, this change has no adverse impact on safety.

DISCUSSION OF CHANGES
ITS SECTION 3.4.11 - Pressurizer Power Operated Relief Valves (PORVs)

LESS RESTRICTIVE

None

REMOVED DETAIL

None

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.11:
## "Pressurizer Power Operated Relief Valves (PORVs)"

## PART 4:

## No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.4.11 - Pressurizer Power Operated Relief Valves (PORVs)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

There are no less restrictive changes for the adoption of this ITS.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.4.11:
# "Pressurizer Power Operated Relief Valves (PORVs)"

## PART 5:

## NUREG-1431
## Annotated to show differences between NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.4.11

This ITS Specification is based on NUREG-1431 Specification No. 3.4.11 as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| WOG-054 R3 | 113 R3 | ELIMINATE SHUTDOWN TO MODE 4 FOR INOPERABLE PORVS | See Next Rev | Not Incorporated | N/A |
| WOG-054 R4 | 113 R4 | ELIMINATE SHUTDOWN TO MODE 4 FOR INOPERABLE PORVS | NRC Rejects: TSTF to Revise | Not Incorporated | N/A |
| WOG-060 R1 | | PORV SR NOTES ADDED | TSTF Review | Not Incorporated | N/A |
| WOG-061 | 151 R0 | PORV OPERABILITY CLARIFICATION | TSTF to Rewrite | Incorporated Operability definition portions only. | T.1 |
| WOG-102 | | SEPARATE CONDITION ENTRY FOR EACH PORV AND BLOCK VALVE | TSTF Review | Not Incorporated | N/A |

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

<3.1.A.4>
<3.1.A.5>

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

<3.1.A.4>
<3.1.A.5>
<DOC M.2>

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTES-----

<DOC A.5> 1. Separate Condition entry is allowed for each PORV.

<DOC A.4> 2. LCO 3.0.4 is not applicable.

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <3.1.A.4> <DOC M.1> | A. One or more PORVs inoperable and capable of being manually cycled. | A.1 Close and maintain power to associated block valve. | 1 hour |
| <DOC M.1> | B. One ~~[or two]~~ PORV~~[s]~~ inoperable and not capable of being manually cycled. | B.1 Close associated block valve~~[s]~~. | 1 hour |
| | | AND | |
| | | B.2 Remove power from associated block valve~~[s]~~. | 1 hour |
| | | AND | |
| | | B.3 Restore PORV~~[s]~~ to OPERABLE status. | 72 hours 7 days X.1 |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| <DOC M.1> C. One block valve inoperable. | C.1 Place associated PORV in manual control. | 1 hour |
| | **AND** | |
| | C.2 Restore block valve to OPERABLE status. | 72 hours 7 days |
| <DOC M.1> D. Required Action and associated Completion Time of Condition A, B, or C not met. | D.1 Be in MODE 3. | 6 hours |
| | **AND** | |
| | D.2 Be in MODE 4. | 12 hours |
| <DOC M.1> E. Two ~~[or three]~~ PORVs inoperable and not capable of being manually cycled. | E.1 Close associated block valves. | 1 hour |
| | **AND** | |
| | E.2 Remove power from associated block valves. | 1 hour |
| | **AND** | |
| | E.3 Be in MODE 3. | 6 hours |
| | **AND** | |
| | E.4 Be in MODE 4. | 12 hours |
| <DOC M.1> F. More than one block valve inoperable. | F.1 Place associated PORVs in manual control. | 1 hour |
| | **AND** | |
| | | (continued) |

ACTIONS

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <DOC M.1> | F. (continued) | F.2 Restore one block valve to OPERABLE status ~~[if three block valves are inoperable]~~. | 2 hours |
| | | ~~AND~~ | |
| | | ~~F.3 Restore remaining block valve(s) to OPERABLE status.~~ | ~~72 hours~~ |
| <DOC M.1> | G. Required Action and associated Completion Time of Condition F not met. | G.1 Be in MODE 3. | 6 hours |
| | | AND | |
| | | G.2 Be in MODE 4. | 12 hours |

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | | FREQUENCY |
|---|---|---|---|
| <DOC A.6> | SR 3.4.11.1 | -------NOTE------- Not required to be met with block valve closed in accordance with the Required Action of Condition B or E. | |
| <Table 4.1-3, Item 15> | | Perform a complete cycle of each block valve. | 92 days |
| <Table 4.1-3, Item 14> | SR 3.4.11.2 | Perform a complete cycle of each PORV. | ~~[18]~~ 24 months |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.11.3 | Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems. | ~~[18]~~ months |
| SR 3.4.11.4 | Verify PORVs and block valves are capable of being powered from emergency power sources. | [18] months |

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are air operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

nitrogen

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

and alternate

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

Electrical power needed to support the

The PORVs, their block valves, and their controls ~~are powered~~ from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

is supplied

is

supplied

design

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure—High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition,

179,000

and automatic reactor control operation

(continued)

BASES

BACKGROUND (continued)

the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

APPLICABLE SAFETY ANALYSES

and alternate

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray ~~is~~ are not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray or auxiliary spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator. (Ref. 2)

The PORVs are ~~used~~ modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical~~.~~ although not required to meet safety limits. By assuming PORV manual actuation, ~~the primary pressure remains below the high pressurizer pressure trip setpoint; thus~~, the DNBR calculation is more conservative. ~~Events that assume this condition include a turbine trip and the loss of normal feedwater (Ref. 2)~~. Insert: B 3.4-51-01

T.1

Pressurizer PORVs satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. ~~The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive leakage.~~ Insert: B 3.4-51-02 Satisfying the LCO helps minimize challenges to fission product barriers.

T.1

(continued)

INSERT: B 3.4-51-01

As such, this actuation is not required to mitigate these events, and PORV automatic operation is not an assumed safety function.

INSERT: B 3.4-51-02

An OPERABLE block valve may be either open, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage.

BASES (continued)

**APPLICABILITY**

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 ~~to minimize challenges to the pressurizer safety valves~~.

Insert: B3.4-52-01

T.1

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

**ACTIONS**

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). ~~The exception for LCO 3.0.4, Note 2,~~ permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status. Testing is not performed in lower MODES.

• This exception to LCO requirements is normally used

because

A.1

may be

associated

is required to

T.1

~~With the~~ PORVs inoperable and capable of being manually cycled, either the PORVs must be restored or the flow path isolated within 1 hour. The block valves ~~should~~ be closed, but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. ~~Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use and do not create a possibility for a~~

Insert: B3.4-52-02

T.1

(continued)

**INSERT: B 3.4-52-01**

for manual actuation to mitigate a steam generator tube rupture event.

**INSERT: B 3.4-52-02**

(e.g., excessive seat leakage). In this condition,

BASES

ACTIONS

A.1 (continued)

~~small break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve.~~ This ~~Condition is only intended to~~ permit operation (A) of the plant ~~for a limited period of time not to exceed~~ until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2). T.1

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, and B.3

If one ~~[or two]~~ PORV~~[s]~~ is inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. (3) The Completion Time~~s~~ of 1 hour ~~are~~ is reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional ~~72 hours~~ 7 days is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in ~~manual control~~ the closed position (i.e., switch in manual control). The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours 7days to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs are not may not be capable of mitigating an overpressure event ~~when placed in manual control~~ if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 72 hours 7days the power will be restored and the PORV restored ~~to OPERABLE status~~. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

T.1

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, ~~maintaining~~ PORV OPERABILITY may be required. See LCO 3.4.12.

E.1, E.2, E.3, and E.4

If more than one PORV is inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time

(continued)

BASES

ACTIONS

E.1, E.2, E.3, and E.4 (continued)

to correct the situation. ~~If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two [or three] PORVs inoperable.~~ If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

F.1, F.2, and ~~F.3~~

If more than one block valve is inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control (i.e, closed position) and restore at least one block valve within 2 hours ~~[and restore the remaining block valve within 72 hours]~~. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

opened and

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the Frequency of 92 days is the ASME Code, Section XI (Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is ~~of~~ importance, (1) because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an ~~otherwise~~ inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is ~~72 hours~~, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status ~~(i.e., completion of the Required Actions fulfills the SR)~~.

that is not capable of being manually cycled

7 days

The Note modifies this SR by stating that it is not required to be met with the block valve closed, in accordance with the Required Action of this LCO.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.

24

~~SR 3.4.11.3~~

~~Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.~~

~~SR 3.4.11.4~~

~~This Surveillance is not required for plants with permanent 1E power supplies to the valves.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.4 (continued)

The Surveillance demonstrates that emergency power can be provided and is performed by transferring power from normal to emergency supply and cycling the valves. The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

1. Regulatory Guide 1.32, February 1977.
2. FSAR, Section ~~[15.2]~~. 14
3. ASME, Boiler and Pressure Vessel Code, Section XI.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.4.11:
# "Pressurizer Power Operated Relief Valves (PORVs)"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.4.11 - Pressurizer Power Operated Relief Valves (PORVs)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change partially incorporates Generic Change TSTF-151, Rev.0 (WOG-61). Specifically, those portions of TSTF-151 that provide clarification in the Bases for the requirements for PORV Operability are incorporated.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

X.1 IP3 ITS differs from NUREG-1431 by extending the allowable out of service time from 72 hours to 7 days for one PORV inoperable and not capable of being manually cycled (loss of redundancy of the manual venting function) (Condition B) and for one block valve inoperable (i.e., not capable of being manually cycled) (Condition C). This change is acceptable because this the AOT for loss of redundancy of the manual venting function used to reduce RCS pressure following a SGTR. During the 7 day AOT, one PORV vent path is still available for venting and both normal and alternate pressurizer spray are typically available to perform the same function.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.4.12:
# "Low Temperature Overpressure Protection (LTOP)"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

# 3.4 REACTOR COOLANT SYSTEM (RCS)

## 3.4.12 Low Temperature Overpressure Protection (LTOP)

LCO 3.4.12

LTOP shall be OPERABLE with no high head safety injection (HHSI) pumps capable of injecting into the RCS and the accumulator discharge isolation valves closed and de-energized, and either of the following:

----------------------------Note----------------------------
LCO 3.4.12.a and LCO 3.4.12.b are not Applicable when average RCS cold leg temperature is ≥ 319°F.
------------------------------------------------------------

a. The Overpressure Protection System (OPS) OPERABLE with two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR;

<u>OR</u>

b. The RCS depressurized with an RCS vent of ≥ 2.00 square inches.

----------------------------NOTES----------------------------
1. Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

2. One HHSI pump may be made capable of injecting into the RCS as needed to support emergency boration or to respond to a loss of RHR cooling.

3. One HHSI pump may be made capable of injecting into the RCS for pump testing for a period not to exceed 8 hours.
------------------------------------------------------------

APPLICABILITY:

Whenever the RHR System is not isolated from the RCS,
MODE 4 when average RCS cold leg temperature is < 319°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more HHSI pump(s) capable of injecting into the RCS . | A.1 Initiate action to verify no HHSI pumps are capable of injecting into the RCS. | Immediately |
| | OR | |
| | A.2.1 Verify RCS is vented with opening ≥ 2.00 square inches. | Immediately |
| | AND | |
| | A.2.2 Verify pressurizer level is ≤ 0%. | Immediately<br>AND<br>Once per 12 hours |
| | AND | |
| | A.2.3 Verify no more than two HHSI pumps are capable of injecting into the RCS. | Immediately<br>AND<br>Once per 12 hours |
| | OR | |
| | A.3.1 Verify RCS is vented with opening greater than or equal to one pressurizer code safety valve flange. | Immediately |
| | AND | |
| | A.3.2 Verify no more than two HHSI pumps are capable of injecting into the RCS | Immediately<br>AND<br>Once per 12 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| B. An accumulator discharge isolation valve not closed and de-energized when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR. | B.1 | Close and de-energize isolation valve for affected accumulator. | 1 hour |
| C. Required Action and associated Completion Time of Condition B not met. | C.1.1 | Increase average RCS cold leg temperature to ≥ 319°F. | 12 hours |
| | <u>AND</u> | | |
| | C.1.2 | Isolate the RHR System from the RCS. | 12 hours |
| | <u>OR</u> | | |
| | C.2 | Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR. | 12 hours |
| D. One required PORV inoperable. | D.1 | Restore required PORV to OPERABLE status. | 7 days |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| E. Two required PORVs inoperable.<br><br>OR<br><br>Required Action and associated Completion Time of Condition C or D not met. | E.1 | Depressurize RCS and establish RCS vent of ≥ 2.00 square inches. | 8 hours |
| | OR | | |
| | E.2 | Increase RCS cold leg temperature to ≥ 319°F. | 8 hours |
| | OR | | |
| | E.3 | Verify pressurizer level, RCS pressure, and RCS injection capability are within limits specified in PTLR for OPS not OPERABLE. | 8 hours<br><br>AND<br><br>Once per 12 hours thereafter |
| F. LTOP inoperable for any reason other than Condition A, B, C, D, or E. | F.1 | Depressurize RCS and establish RCS vent of ≥ 2.00 square inches. | 8 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.12.1 | Verify no HHSI pumps are capable of injecting into the RCS. | 12 hours |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.12.2 | Verify each accumulator discharge isolation valve is closed and de-energized;<br><br>OR<br><br>Verify each accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. | 12 hours<br><br><br><br>12 hours |
| SR 3.4.12.3 | -----------------NOTE-----------------<br>Only required to be met when complying with LCO 3.4.12.b.<br>--------------------------------------<br><br>Verify RCS vent ≥ 2.00 square inches established. | 12 hours for unlocked open vent valve(s)<br><br>AND<br><br>31 days for locked open vent valve(s) |
| SR 3.4.12.4 | -----------------NOTE-----------------<br>Only required to be met when complying with LCO 3.4.12.a.<br>--------------------------------------<br><br>Perform CHANNEL CHECK of Overpressure Protection (OPS) instrument channels. | 24 hours |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.12.5 | Verify PORV block valve is open for each required PORV. | 72 hours |
| SR 3.4.12.6 | -------------------NOTE--------------------- Not required to be met until 12 hours after decreasing RCS average cold leg temperature to < 319°F. ------------------------------------------- Perform a COT on each required PORV, excluding actuation. | 24 months |
| SR 3.4.12.7 | Perform CHANNEL CALIBRATION for each required OPS channel as follows: | |
| | a. OPS actuation channels; and | 18 months |
| | b. RCS pressure and temperature instruments. | 24 months |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.12.8 | -------------------NOTES-------------------<br>1. Not required to be met when average RCS cold leg temperature is ≥ 319°F.<br><br>2. Not required to be met if SR 3.4.12.9 is met.<br>------------------------------------------<br><br>Verify each of the following conditions are satisfied prior to starting any RCP:<br><br>a. Secondary side water temperature of the hottest steam generator (SG) is less than or equal to the coldest RCS cold leg temperature; and<br><br>b. RCS makeup is less than or equal to RCS losses; and<br><br>c. Steam generator pressure is not decreasing; and<br><br>d.1 Overpressure Protection System (OPS) is OPERABLE;<br><br>OR<br><br>d.2.1 RCS pressure less than nominal OPS setpoint specified in the PTLR; and<br><br>d.2.2 Pressurizer level, RCS pressure, and RCS injection capability are within limits specified in PTLR for OPS not OPERABLE. | Within 15 minutes prior to starting any RCP |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.12.9 | ---NOTES---<br>1. Not required to be met when average RCS cold leg temperature is ≥ 319°F.<br><br>2. Not required to be met if SR 3.4.12.8 is met.<br>---<br><br>Verify each of the following conditions are satisfied prior to starting any RCP:<br><br>a. Secondary side water temperature of the hottest steam generator is ≤ 64°F above the coldest RCS cold leg temperature; and<br><br>b. RCS makeup is less than or equal to RCS losses; and<br><br>c. Overpressure Protection System (OPS) is OPERABLE; and<br><br>d. Pressurizer level is ≤ 73%; and<br><br>e. Coldest RCS cold leg temperature is within limits specified in PTLR for RCP start with OPS OPERABLE and SG temperature greater than RCS cold leg temperature | Within 15 minutes prior to starting any RCP |

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.12 Low Temperature Overpressure Protection (LTOP)

BASES

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BACKGROUND

LTOP is established to limit RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown because a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

When the RHR System is isolated from the RCS, the RHR System is protected from overpressure by two spring loaded relief valves (SI-733A and SI-733B). When the RHR System is not isolated from the RCS, the RHR System is protected from overpressure by spring loaded relief valve (i.e., AC-1836) which has sufficient capacity to accommodate all 3 charging pumps. However, this relief valve does not have sufficient capacity to ensure that the RHR system does not exceed design pressure limits during a mass addition resulting from an inadvertent injection of one or more high head safety injection (HHSI) pumps. Therefore, LTOP requirements are

BASES

BACKGROUND (Continued)

used to protect the RHR System whenever the RHR System is not isolated from the RCS.

This LCO provides RCS overpressure protection by limiting maximum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability is achieved by not permitting any High Head Safety Injection (HHSI) pumps to be capable of injection into the RCS and isolating the accumulators. The pressure relief capacity requires either two redundant power operated relief valves (PORVs) or a depressurized RCS and an RCS vent of sufficient size. One PORV or the open RCS vent is sufficient to provide overpressure protection to terminate an increasing pressure event. Alternately, if redundant PORVs are not Operable or an RCS vent cannot be established, LTOP protection may be established by limiting the pressurizer level to within limits specified in the PTLR consistent with the number of charging pumps and number of high head safety injection (HHSI) pumps capable of injecting into the RCS. This approach is acceptable because pressurizer level can be maintained such that it will either accommodate any anticipated pressure surge or allow operators time to react to any unanticipated pressure surge. When pressurizer level is used to satisfy LTOP requirements, operator action is assumed to terminate the unplanned HHSI pump injection within 10 minutes.

With high pressure coolant input capability limited, the ability to create an overpressure condition by coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. There is no restriction on the status of charging pumps when LTOP is established using either a PORV or an RCS vent. If conditions require the use of more than one HHSI pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions. Charging pumps and low pressure injection systems are available to provide makeup even when LTOP requirements are applicable.

BASES

BACKGROUND (Continued)

When configured to provide low temperature overpressure protection, the PORVs are part of the Overpressure Protection System (OPS). LTOP for pressure relief can consist of either the OPS (two PORVs with reduced lift settings), or a depressurized RCS and an RCS vent of sufficient size. Two PORVs are required for redundancy. One PORV has adequate relieving capability to keep from overpressurization for the required coolant input capability.

PORV Requirements

The Overpressure Protection System (OPS) provides the low temperature overpressure protection by controlling the Power Operated Relief Valves (PORVs) and their associated block valves with pressure setpoints that vary with RCS cold leg temperature. Specifically, cold leg temperature signals from three RCS loops are supplied to three associated function generators that calculate the maximum RCS pressures allowed at those temperatures. The maximum RCS pressure limits at any RCS temperature correspond to the 10 CFR 50, Appendix G, limit curve maintained in the Pressure and Temperature Limits Report and are used as the OPS pressure setpoint. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

In addition to generating the OPS pressure setpoint, the same cold leg temperature signals are used to "arm" the OPS when RCS temperature falls below the temperature at which low temperature overpressure protection is required (319°F). Each PORV opens when a two-out-of-two (temperature and pressure) coincidence logic is satisfied. OPS is "armed" when RCS temperature falls below the temperature that satisfies one half of the two-out-of-two (temperature-pressure) coincidence logic. When OPS is enabled, the PORVs will open if RCS pressure exceeds the calculated pressure setpoint that varies with RCS temperature. The PORV block valves open when the RCS temperature falls below the OPS arming temperature. Note that the control switches for the PORV and PORV block valves must be in the AUTO position and the OPS states links closed for OPS signals to actuate the PORVs.

BASES

BACKGROUND (Continued)

Three channels of RCS cold leg temperature are used in the two-out-of-three coincidence logic to satisfy the temperature portion of the two-out-of-two (temperature and pressure) coincidence logic for each PORV. Three channels of RCS pressure are used in a two-out-of-three coincidence logic to satisfy the pressure portion of the two-out-of-two (temperature-pressure) coincidence logic for each PORV. Use of a two-out-of-three coincidence logic for pressure and for temperature ensures that a single failure will not cause or prevent an OPS actuation. Use of two PORVs, each with adequate relieving capability to prevent overpressurization, ensures that a single failure will not prevent an OPS actuation.

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

Multiple methods exist for establishing the required RCS vent capacity including removing or blocking open a PORV and disabling its block valve in the open position. An RCS vent of ≥ 2.00 square inches when no HHSI pump is capable of injecting into the RCS; or, an RCS vent with opening greater than or equal to one pressurizer code safety valve flange and up to two HHSI pumps capable of injecting into the RCS will satisfy LTOP requirements because either configuration ensures pressure limits are not exceed during a transient. Alternately, an RCS vent of ≥ 2.00

BASES

BACKGROUND (Continued)

square inches coupled with a pressurizer level ≤ 0% and up to two HHSI pumps capable of injecting into the RCS will satisfy LTOP requirements because it ensures a minimum of 10 minutes for operator action before pressure limits are exceeded during a transient. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, with RCS cold leg temperature exceeding 411°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At 319°F and below, overpressure prevention falls to two OPERABLE PORVs in conjunction with the Overpressure Protection System (OPS) or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability. Alternately, if redundant PORVs are not Operable, Low Temperature Overpressure protection may be maintained by limiting the pressurizer level to within limits specified in the PTLR consistent with the number of charging pumps and number of high head safety injection (HHSI) pumps capable of injecting into the RCS. This approach is acceptable because pressurizer level can be established to either accommodate any anticipated pressure surge or allow operators time to react to any unanticipated pressure surge.

When the RCS temperature is greater than the LTOP arming temperature (i.e., ≥ 319°F) but below the minimum temperature at which the pressurizer safety valves lift prior to violation of the 10 CFR 50, Appendix G, limits (i.e., ≤ 411°F), administrative controls in the Technical Requirements Manual (TRM) (Ref. 4) are used to limit the potential for exceeding 10 CFR 50, Appendix G, limits. These administrative controls may include operating with a bubble in the pressurizer and/or otherwise limiting plant time or activities when the RCS temperature is in the specified range. The use of administrative controls to govern operation above the LTOP arming temperature

BASES

APPLICABLE SAFETY ANALYSES (continued)

but below the minimum temperature at which the pressurizer safety valves lift prior to violation of the 10 CFR 50, Appendix G, limits is consistent with the guidance provided in Generic Letter 88-011, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations (Ref.2). GL 88-011 states that automatic, or passive, protection of the P-T limits will not be required but administratively controlled when in the upper end of the 10 CFR 50, Appendix G, temperature range.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, LTOP must be re-evaluated to ensure its functional requirements can still be met using the OPS (PORVs) method or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Ref. 3 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur. This is accomplished by the following:

a. Rendering all HHSI pumps incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions or maintaining accumulator pressure less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR; and

c. Disallowing start of an RCP unless conditions are established that ensure a RCP pump start will not cause a pressure excursion that will exceed LTOP limits. Required conditions for starting a RCP when LTOP is required include a combination of primary and secondary water temperature differences and Overpressure Protection System (OPS) status or pressurizer level. Meeting the LTOP RCP starting surveillances ensures that theses conditions are satisfied prior to a RCP pump start.

The Ref. 3 analyses demonstrate that either one PORV or the depressurized RCS and RCS vent can maintain RCS pressure below limits when no HHSI pump is capable of injecting into the RCS. This assumes an RCS vent of ≥ 2.00 square inches. The same protection can be provided when up to two HHSI pumps are capable of injecting into the RCS assuming an RCS vent with opening greater than or equal to one code pressurizer safety valve flange. Alternately, LTOP requirements can be satisfied by various combinations of pressurizer level, RCS pressure, and RCS injection capability (i.e., maximum number of HHSI pumps and/or charging pumps) shown in the PTLR. These combinations of pressurizer level, RCS pressure, and RCS injection capability satisfy LTOP requirements by ensuring a minimum of 10 minutes for operator action to terminate an unplanned event prior to

BASES

APPLICABLE SAFETY ANALYSES (continued)

exceeding maximum allowable RCS pressure. None of the analyses addressed the pressure transient need from accumulator injection, therefore, when RCS temperature is low, the LCO also requires the accumulator isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

If the accumulators are isolated and not depressurized, then the accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

Fracture mechanics analyses established the temperature of LTOP Applicability at 319°F.

The consequences of a loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6) requirements by having ECCS OPERABLE in accordance with requirements in LCO 3.5.3, ECCS-Shutdown.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient with HHSI not injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met. The OPS setpoint is based on a comparative analysis of Reference 3, with allowances for metal/fluid temperature differences, static head due to elevation differences, and dynamic head from the operation of the reactor coolant pumps and RHR pumps.

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron

BASES

APPLICABLE SAFETY ANALYSES (continued)

fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 1.4 square inches is capable of mitigating the allowed LTOP overpressure transient assuming no HHSI pump and no accumulator injects into the RCS. The LCO limit for an RCS vent is conservatively established at 2.00 square inches. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, maintaining RCS pressure less than the maximum pressure on the P/T limit curve. An RCS vent with opening greater than or equal to one pressurizer code safety valve flange and up to two HHSI pumps capable of injecting into the RCS will satisfy LTOP requirements because it ensures pressure limits are not exceed during a transient. An RCS vent of ≥ 2.00 square inches coupled with a pressurizer level ≤ 0% and up to two HHSI pumps capable of injecting into the RCS will satisfy LTOP requirements because it ensures a minimum of 10 minutes for operator action before pressure limits are exceeded during a transient.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

LTOP satisfies Criterion 2 of 10 CFR 50.36.

BASES

LCO

This LCO requires that LTOP is OPERABLE. LTOP is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires that no HHSI pumps be capable of injecting into the RCS and all accumulator discharge isolation valves closed and de-energized if accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two OPERABLE PORVs configured as part of an OPERABLE Overpressure Protection System (OPS); or

b. A depressurized RCS and an RCS vent.

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits. The OPS is OPERABLE for LTOP when there are three OPERABLE RCS pressure channels and three OPERABLE RCS temperature channels. The OPS is still OPERABLE when an inoperable RCS pressure or temperature channel is in the tripped condition.

An RCS vent is OPERABLE when open with an area of ≥ 2.00 square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable whenever the RHR System is not isolated from the RCS to protect the RHR system piping. When average RCS cold leg temperatures are ≥ 319°F, RHR system piping is adequately protected by making the accumulators and all HHSI

BASES

APPLICABILITY (continued)

pumps incapable of injecting into the RCS. Therefore, a Note in the LCO specifies that requirements for the OPS System and/or an RCS vent are not Applicable when average RCS cold leg temperature is ≥ 319°F.

This LCO is applicable to provide protection for the RCS pressure boundary in MODE 4 when average RCS cold leg temperature is < 319°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 319°F. When the reactor vessel head is off, overpressurization cannot occur. Although LTOP is not Applicable when the RCS temperature is greater than the LTOP arming temperature (i.e., ≥ 319°F) but below the minimum temperature at which the pressurizer safety valves lift prior to violation of the 10 CFR 50, Appendix G, limits (i.e., ≤ 411°F), administrative controls in the Technical Requirements Manual (TRM) (Ref. 4) are used to limit the potential for exceeding 10 CFR 50, Appendix G, limits.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 319°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

The Applicability is modified by three Notes, Note 1 states that accumulator isolation is only required when the accumulator pressure is more than the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

Note 2 ensures that LCO 3.4.12 will not prohibit a HHSI pump being energized and aligned to the RCS as needed to support emergency boration or to respond to a loss of RHR cooling.

BASES

APPLICABILITY (continued)

Note 3 specifies that one HHSI pump may be made capable of injecting into the RCS for a period not to exceed 8 hours to perform pump testing. During testing, administrative controls are used to ensure that HHSI testing will not result in exceeding RCS or RHR system pressure limits.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, A.3.1 and A.3.2

When one or more HHSI pumps are capable of injecting into the RCS, LTOP assumptions regarding limits on mass input capability may not be met. Therefore, immediate action is required to limit injection capability consistent with the LTOP analysis assumptions and the existing combination of pressurizer level and RCS venting capacity. Required Action A.1 requires restoration with LCO requirements. Required Actions A.2 and A.3 require verification and periodic re-verification that alternate LTOP configurations are met. The Completion Times of immediately reflects the urgency that one of the acceptable LTOP configurations is established as soon as possible.

B.1, C.1 and C.2

To be considered isolated, an accumulator must have its discharge valves closed and the valve power supply breakers fixed in the open position.

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in the next 12 hours. By

BASES

ACTIONS

B.1, C.1 and C.2 (continued)

increasing the RCS temperature to ≥ 319°F, an accumulator pressure of 700 psig cannot exceed the LTOP limits if the accumulators are injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection. Additionally, the RHR System must be isolated from the RCS to protect RHR piping from a potential mass addition event.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

D.1

When average RCS cold leg temperature is < 319°F, with one required PORV inoperable, the PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the PORVs is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

E.1

When both required PORVs are inoperable or the Required Action and associated Completion Time of Condition C or D is not met, an alternate method of low temperature overpressure protection must be established within 8 hours. The acceptable alternate methods of LTOP include the following:

a. Depressurize the RCS and establish an RCS vent path; or

b. Increase average RCS cold leg temperatures to ≥ 319°F; or

If the option selected is to depressurize the RCS and establish an RCS vent path, the vent must be sized ≥ 2.00 square inches to ensure that the flow capacity is greater than that required for

BASES

ACTIONS

E.1 (continued)

c. Establish a combination of pressurizer level, RCS pressure, and RCS injection capability within limits specified in PTLR for OPS not OPERABLE. This combination will ensure at least 10 minutes for operator intervention to prevent overpressurization following a transient.

the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

F.1

If LTOP requirements are not met for reasons other than Conditions A, B, C, D or E, LTOP requirements must be re-established by depressurizing the RCS and establishing an RCS vent of ≥ 2.00 square inches within 8 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, all HHSI pumps are verified incapable of injecting into the RCS. Additionally, the accumulator discharge isolation valves are verified closed and locked out or the accumulator pressure less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

The HHSI pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2 (continued)

out under administrative control. Other methods may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in Trip Pullout and at least one valve in the discharge flow path being closed.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.12.3

The RCS vent of ≥ 2.00 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve, PORV, or Manway Cover fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12.b.

SR 3.4.12.4

Performance of the CHANNEL CHECK of the Overpressure Protection System (OPS) RCS pressure and temperature channels every 24 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.4 (continued)

value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels. This SR is required only when LCO 3.4.12.a is used to establish LTOP protection.

SR 3.4.12.5

The PORV block valve opens automatically when RCS cold leg temperature is below the OPS arming temperature; however, the valves must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve may be remotely verified open in the control room. This Surveillance is performed only if the PORV is being used to satisfy LCO 3.4.12.a.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation. If closed, the block valve

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.5 (contdinued)

must be de-energized to prevent the valve from re-opening automatically.

The 72 hour Frequency is considered adequate because the PORV block valves are opened automatically by the OPS when below the OPS arming temperature if the valve control is positioned to auto and other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.6

Performance of a COT is required within 12 hours after decreasing RCS temperature to < 319°F and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

The 31 day Frequency considers the demonstrated reliability of the Overpressure Protection System and the PORVs.

A Note has been added indicating that this SR is required to be met 12 hours after decreasing RCS cold leg temperature to < 319 °F. The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.

SR 3.4.12.7

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months. Performance of a CHANNEL CALIBRATION of RCS pressure and temperature instruments that support the Overpressure Protection System is required every 24 months. These calibrations verify both the OPS and PORV function and ensure the OPERABILITY of the whole channel so that

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.7 (continued)

it responds and the valve opens within the required range and accuracy to known input.

SR 3.4.12.8 and SR 3.4.12.9

The RCP starting prerequisites must be satisfied prior to starting or jogging any reactor coolant pump (RCP) when low temperature overpressure protection is required. The RCP starting prerequisites prevent an overpressure event due to thermal transients when an RCP is started. Plant conditions prior to the RCP start determines whether SR 3.4.12.8 or SR 3.4.12.9 must be satisfied prior to starting any RCP.

The principal contributor to an RCP start induced thermal and pressure transient is the difference between RCS cold leg temperatures and secondary side water temperature of any SG prior to the start of an RCP. The RCP starting prerequisites vary depending on plant conditions but include the following: reactor coolant temperature relative to the LTOP enable temperature; secondary side water temperature of the hottest SG relative to the temperature of the coldest RCS cold leg temperature; and, status of the Overpressure Protection System (OPS). When the OPS is inoperable, additional compensatory requirements are required including limits for the pressurizer level and RCS pressure and temperature. When a pressurizer level is specified as a requirement, the level specified is sufficient to prevent the RCS from going water solid for 10 minutes which is sufficient time for operator action to terminate the pressure transient.

SR 3.4.12.8 is used if secondary side water temperature of the hottest steam generator (SG) is less than or equal to the coldest RCS cold leg temperature. SR 3.4.12.9 is more restrictive and is used if the secondary side water temperature of the hottest steam generator is ≤ 64°F above the coldest RCS cold leg temperature.

RCP starting is prohibited if the hottest steam generator is > 64°F above RCS cold leg temperature or if neither of the RCP starting prerequisites SRs can be satisfied.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.8 and SR 3.4.12.9 (continued)

The steam generator temperature may be measured using the Control Room instrumentation or, as a backup, from a contact reading off the steam generator's shells. Pressurizer level may be determined using control room instrumentation or alternate methods.

The FREQUENCY of the RCP starting prerequisites SRs is Within 15 minutes prior to starting any RCP. This means that each of the required verifications must be performed within 15 minutes prior to the pump start and must be met at the time of the pump start.

SR 3.4.12.8 and SR 3.4.12.9 are each modified by two Notes. Note 1 specifies that these SRs are required as a condition for pump starting only when the RCS is below the LTOP arming temperature. Note 2 specifies that meeting either SR 3.4.12.8 or SR 3.4.12.9 ensures that pump starting prerequisites are met.

BASES

REFERENCES

1. 10 CFR 50, Appendix G.

2. Generic Letter 88-011, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations.

3. IP3 Low Temperature Overpressurization System Analysis Final Report, August 24, 1984, in conjunction with ASME Code Case N-514, Low Temperature Overpressure Protection, February 12, 1992.

4. IP3 Technical Requirements Manual.

5. 10 CFR 50, Section 50.46.

6. 10 CFR 50, Appendix K.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.12:
## "Low Temperature Overpressure Protection (LTOP)"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.1-2 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-3 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-5 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-6 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-8 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-9 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-10 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-11(F 3.1.A-1) | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-12(F 3.1.A-2) | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-13(F 3.1.A-3) | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-14(F 3.1.A-4) | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-15(F 3.1.A-5) | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |

# Indian Point 3
# Improved Technical Specifications (ITS)
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# Technical Specification 3.4.12:
# "Low Temperature Overpressure Protection (LTOP)"

| | | | | | |
|---|---|---|---|---|---|
| **3.1-16(F 3.1.A-6)** | **179** | **179** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **3.3-5a** | **179** | **179** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **T 3.5-3(3)** | **151** | **151 TSCR 96-124** | **IPN 96-124** | AOT for ESF Initiation Instrumentation (Needs Supplement) | **Incorporated** |
| **T 4.1-1(5)** | **169 TSCR 98-043** | **169 TSCR 98-043** | **IPN 98-043** | Instrument Channel Surveillance Intervals Extended to 24 Months | **Incorporated** |

SEE ITS 3.4.7 ITS 3.4.8

d. When the reactor coolant system $T_{avg}$ is less than 200°F, but not in the refueling operation condition, and as permitted during special plant evolutions, at least one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation. This RHR pump may be out of service for up to 1 hour provided no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

SEE ITS 3.4.4

e. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

f. The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation.

g. If the requirements of 3.1.A.1.e and 3.1.A.1.f above cannot be satisfied, the reactor shall be brought to the hot shutdown condition within 1 hour.

SR 3.4.12.8.b SR 3.4.12.9.b

h. A reactor coolant pump (RCP) may not be started ~~(or jogged)~~ A.5 when ~~the RCS cold leg temperature ($T_{cold}$) is at or below 319°F~~ A.4 unless RCS make up is not in excess of RCS losses, and one of the following requirements is met: 15 minutes prior to starting any RCP M.4

SR 3.4.12.8.a SR 3.4.12.8.c SR 3.4.12.8.d.1

(1) The OPS is operable, steam generator hottest A.1 pressure is not decreasing, and the temperature of each steam generator is less than or equal to the coldest $T_{cold}$;

Note 2 to SR 3.4.12.8 and SR 3.4.12.9 → or

SR 3.4.12.9.a SR 3.4.12.9.c SR 3.4.12.9.d SR 3.4.12.9.e

(2) The OPS is operable, the temperature of the hottest steam generator exceeds the coldest $T_{cold}$ by no more than 64°F, pressurizer level is at or below 73 percent, and $T_{cold}$ is as per Figure 3.1.A-1 PTLR LA.1

Note 2 to SR 3.4.12.8 and SR 3.4.12.9 → or

SR 3.4.12.8.c SR 3.4.12.8.a SR 3.4.12.8.d.2.1 SR 3.4.12.8.d.2.2

(3) With OPS inoperable, steam generator hottest A.1 pressure is not decreasing, the temperature of each steam generator is less than or equal to the coldest $T_{cold}$, ~~pressurizer level is at or below 73 percent~~, and the RCS pressure does not ~~exceed that given by Fig. 3.1.A-2~~. The pressurizer level must be further restricted per Figures 3.1.A-5 and 3.1.A-6 ~~if operation below 319°F exceeds 8 hours~~ M.3

OPS setpoint in PTLR LA.1

M.3



Amendment ~~48, 53, 67, 84, 93, 121,~~ 179

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Add LCO 3.4.12, accumulators isolated
LCO 3.4.12, Note 1
Add Condition B and associated Req. Act.
Add SR 3.4.12.2
M.1

Add SR 3.4.12.1 M.2

Add SR 3.4.12.3 M.6

Add SR 3.4.12.5 M.7 A.6

SEE CTS RELOCATED

7. REACTOR VESSEL HEAD VENTS

Whenever the reactor coolant system is above 350°F, two reactor vessel head vent paths consisting of two valves in series with power available from emergency buses shall be OPERABLE.

a. If one of the above reactor vessel head vent paths is inoperable, startup and/or power operation may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path. Restore the inoperable vent path to operable status within 90 days, or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

b. With both reactor vessel head vent paths inoperable restore one vent path to operable status within 7 days or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

LCO 3.4.12 ~~8.~~ ~~OVERPRESSURE PROTECTION SYSTEM (OPS)~~ LTOP A.3

Applicability a. When the RCS temperature is below 319°F, A.4

LCO 3.4.12.a ~~(1)~~ the OPS shall be ~~armed and~~ operable. Both OPS PORVs shall have lift settings not to exceed those given in ~~Figure 3.1.A-2,~~ or PTLR A.6 LA.1

LCO 3.4.12.b ~~(2)~~ the RCS must be vented with an equivalent opening of 2.00 square inches.

Req. Act D.1 b. The requirements of 3.1.A.8.a may be modified to allow one PORV ~~and/or its series block valve~~ to be inoperable for a maximum of seven (7) consecutive days. A.6

Conditions E and F c. If the requirements of 3.1.A.8.a or 3.1.A.8.b cannot be met, then one of the following actions shall be completed within 8 hours.

Req. Act E.1 (1) The RCS must be depressurized and vented with an equivalent opening of at least 2.00 square inches;

Or

Req. Act E.2 (2) The RCS must be heated in accordance with the restrictions of Specification 3.1.A.1.h(3) and maintained above 411°F; 319°F LA.4

Or

Req. Act E.3 (3) Restrict pressurizer level as per the curves on Figures 3.1.A-5 and 3.1.A-6. PTLR Reverify every 12 hrs M.5 LA.1



Amendment No. ~~63~~, ~~67~~, ~~121~~, 179

Add Condition F and Req. Act F.1 A.9

d. ~~In the event the PORV's or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2.j within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.~~

A.8

A.1

~~The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.~~

~~The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal off possible RCS leakage paths.~~

~~Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor vessel head vent path ensures that capability exists to perform this function.~~

~~The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve power supply or control system does not prevent isolation of the vent path.~~

~~The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.~~

~~The OPS is designed to relieve the RCS pressure for certain unlikely incidents to prevent the peak RCS pressure from exceeding the limits established in Reg. Guide 1.99, Revision 2. The OPS is considered to be operable when the minimum number of required channels (per Table 3.5-3) are available to open the PORVs upon receipt of a high pressure signal which is based upon RCS $T_{cold}$, as shown in Figure 3.1.A-2. (The happy face icon contained on this and other Technical Specification figures indicates the side of the applicable curve in which operation is permissible. Conversely, the sad face icon indicates the side of the applicable curve in which operation is prohibited.) The OPS setpoint is based upon a comparative analysis of References 5 and 9, with allowances for metal/fluid temperature differences (as described below) and for the static head due to elevation differences and dynamic head effect of the operation of the reactor coolant and RHR pumps. "Arming" means that the motor operated block valve (MOV) is in the open position. This can be accomplished either automatically by the OPS when the RCS temperature is less than or equal to 319°F or manually by the control room operator.~~

A.1

The start of an RCP is allowed when the steam generators' temperature does not exceed the RCS and the OPS is operable. During all modes of operation, the steam generator temperature may be measured using the Control Room instrumentation or, as a backup, from a contact reading off the steam generator's shells.

Most start-ups will satisfy these requirements as provided in Specification 3.1.A.1.h (1). In order to allow start of an RCP when the steam generators are hotter than the RCS, requirements for a pressurizer bubble (gas or steam) are developed (technical specification value for pressurizer level includes an allowance for instrument uncertainty). During this Heat Input initiation event the RCS fluid temperature rise is considerably more rapid than the reactor vessel metal temperature rise. Since OPS utilizes a setpoint curve (Figure 3.1.A-2) and the temperature measured is the fluid temperature, and not the reactor vessel metal, it is necessary to shift to the right the OPS setpoint curve by 50°F to ensure the pressure does not exceed the allowable values for the vessel. For the conditions when the OPS is inoperable, additional requirements are developed for the pressurizer bubble, RCS pressure and temperature.

Due to the rate of energy transferred to the RCS, when the RCP is started, the resultant rate of temperature rise and the pressure increase are strongly dependent on the temperature difference between the RCS and the steam generators. The presence of a pressurizer bubble provides for a more moderate pressure increase. The bubble size is sufficient to prevent the RCS from going water solid for 10 minutes during which time operator action will terminate the pressure transient. Pressurizer level refers to indicated level and includes instrument uncertainty. The preventive measures for a Mass Input initiating event (i.e., up to three charging pumps and/or one SI pump) as well as the Heat Input initiating event are described in References (3), (4) and (5). (Also refer to Specifications 3.3.A.8, 3.3.A.9, and 3.3.A.10.) The OPS need not be operable when the RCS temperature is less than 319°F if the RCS is depressurized and vented with an equivalent opening of at least 2.00 square inches. One PORV, blocked fully open, also satisfies this vent area requirement. This opening is adequate to relieve the worst case analyzed. It should be noted that the analysis of record (Reference 5) is based upon a minimum vent area of 1.4 square inches, which for the sake of conservatism has been rounded up to 2.00 square inches.

The OPS enable temperature of 319°F permits the performance of an RCS hydrostatic test (see Fig. 4.3-1) without activating the OPS.

Upon OPS inoperability, the RCS may be heated above 411°F. This temperature is that value for which the RCS heatup and cooldown curves (Figures 3.1-1 and 3.1-2) permit pressurization to the setting of the pressurizer safety valves. Accordingly, with an inoperable OPS and an RCS temperature above 411°F, the pressurizer safety valves will preclude violation of the 10 CFR 50, Appendix G, curves. In addition, the OPS need not be operable upon satisfying the conditions of Specification 3.1.A.8.c(3) which requires the presence of a pressurizer bubble to preclude RCS overpressurization during inadvertent mass inputs. Specification 3.1.A.8.c(3) also places restrictions on the number of charging and SI pumps capable of feeding the RCS (see Specifications 3.3.A.8, 3.3.A.9, and 3.3.A.10). Any pump can be

A.1

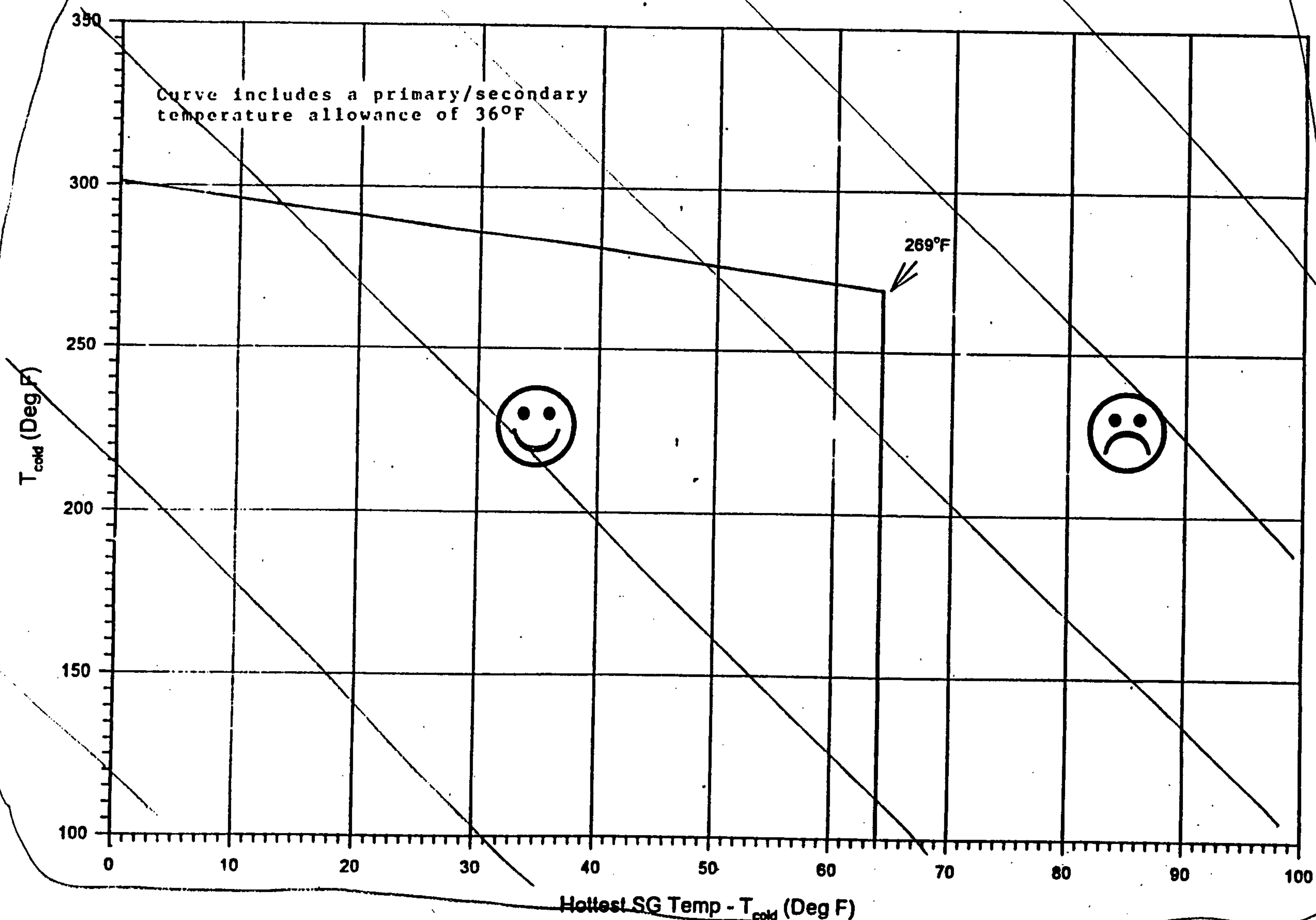
rendered incapable of feeding the RCS if, for example, its switch is in the trip pull-out position, or if at least one valve in the flow path to the RCS is closed and locked (if manual) or de-energized (if motor operated). This section has also been revised in accordance with the results of tests conducted on the capsule T, Y, and Z specimens (References 6, 7 and 8).

References


1) FSAR Section 14.1.6
2) FSAR Section 14.1.8
3) Letter dated 10/25/78 "Summary of Changes to IP-3 Plant Operating Procedures in Order to Preclude RCS Overpressurization"
4) Letter dated 2/28/76 "Conceptual Design of the Reactor Coolant Overpressure Protection System" and response to NRC questions.
5) IP-3 Low Temperature Overpressurization Protection System Analysis, NYPA Report dated 8/24/84.
6) WCAP-9491 "Analysis of Capsule T from IP-3 Reactor Vessel Radiation Surveillance Program", J.A. Davidson, S.L. Anderson, W.T. Kaiser, April 1979.
7) WCAP-10300-1, "Analysis of Capsule Y from the Power Authority of the State of New York Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, S.L. Anderson, March 1983.
8) WCAP-11815, "Analysis of Capsule Z from the New York Power Authority Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, S.L. Anderson, L. Albertin, March 1988.
9) ASME Code Case N-514, "Low Temperature Overpressure Protection," February 12, 1992.

FIG 3.1.A-1

Secondary Side Limitations for RCP Start With
Secondary Side Hotter Than Primary, 13.3 EFPY

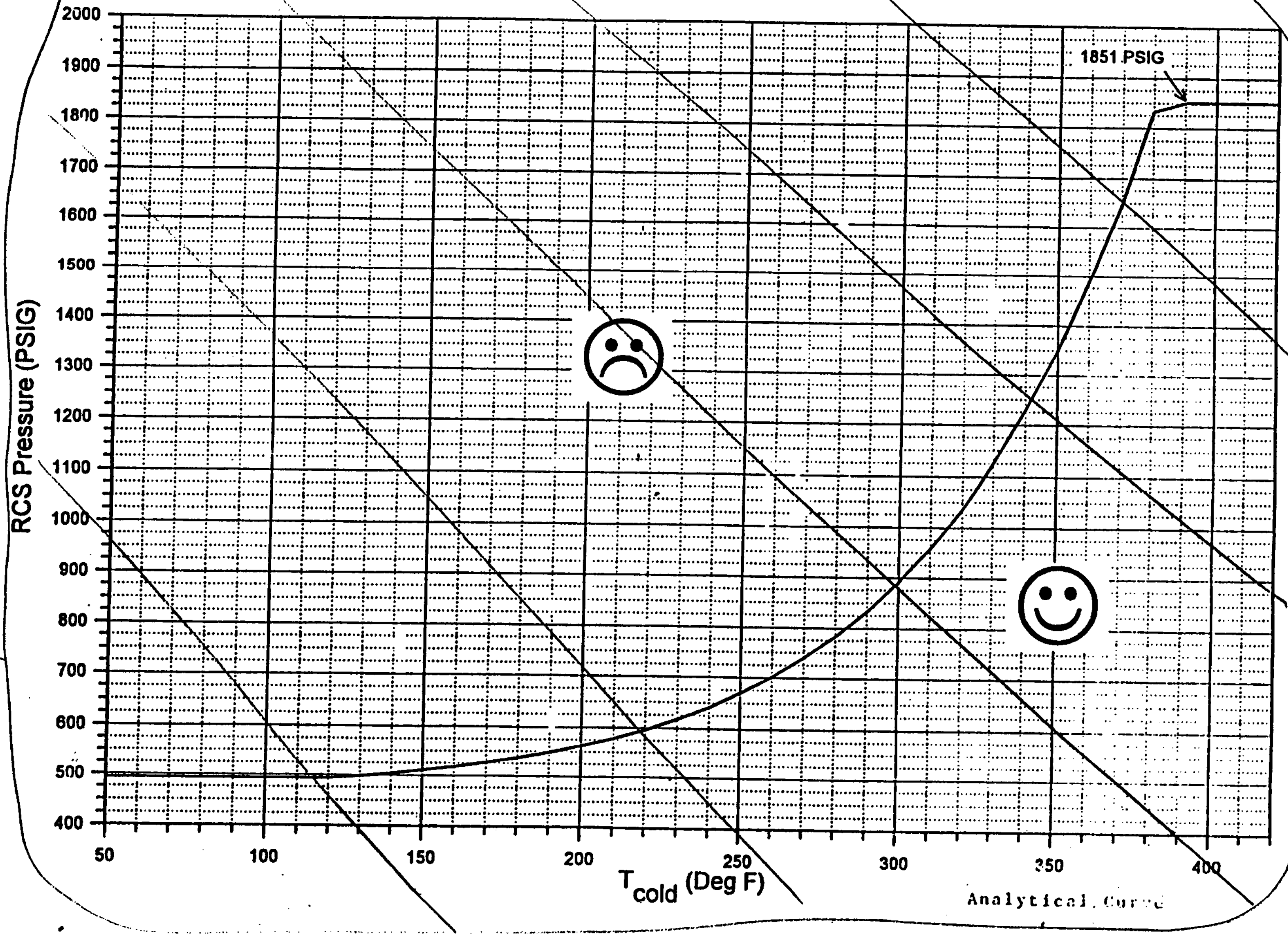



LA.1

ITS 3.4.12

Fi   e 3.1.A-2

Maximum Allowable Nominal PORV Setpoint for the
Low Temperature Overpressure Protection System (OPS), 13.3 EFPY




LA.1

ITS 3.4.12



Amendment No. 67, 121, 179

FIGURE 3.1.A-3

DELETED

A.1

FIGURE 3.1.A-4

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A.1

FIGURE 3.1.A-5

Pressurizer Limitations for OPS Inoperable
(Up to one charging pump capable of feeding RCS), 13.3 EFPY

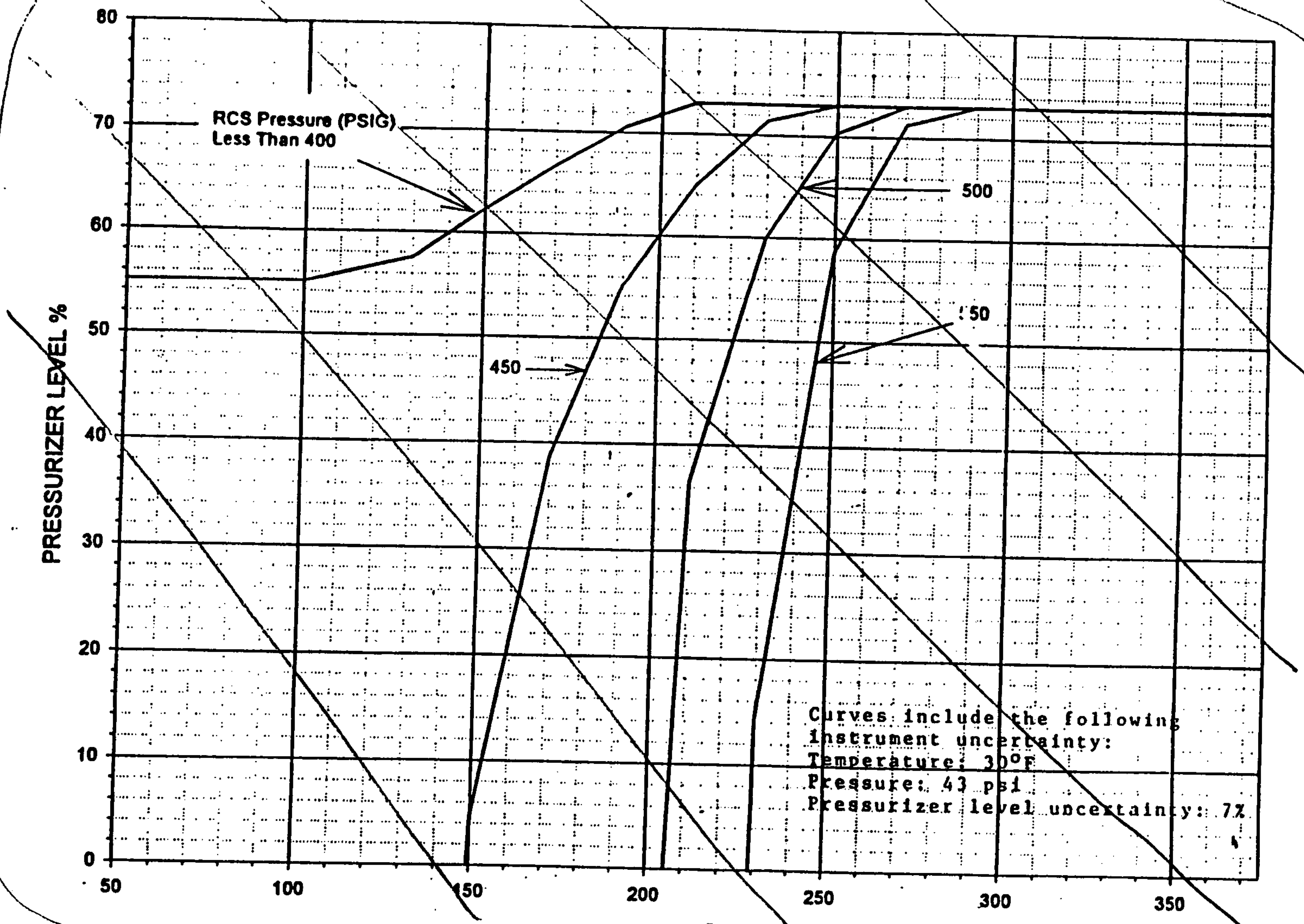


Figure Applicable to 13.3 EFPY

Curves represent maximum allowable pressurizer level for the conditions defined.

LA.1

ITS 3.4.12



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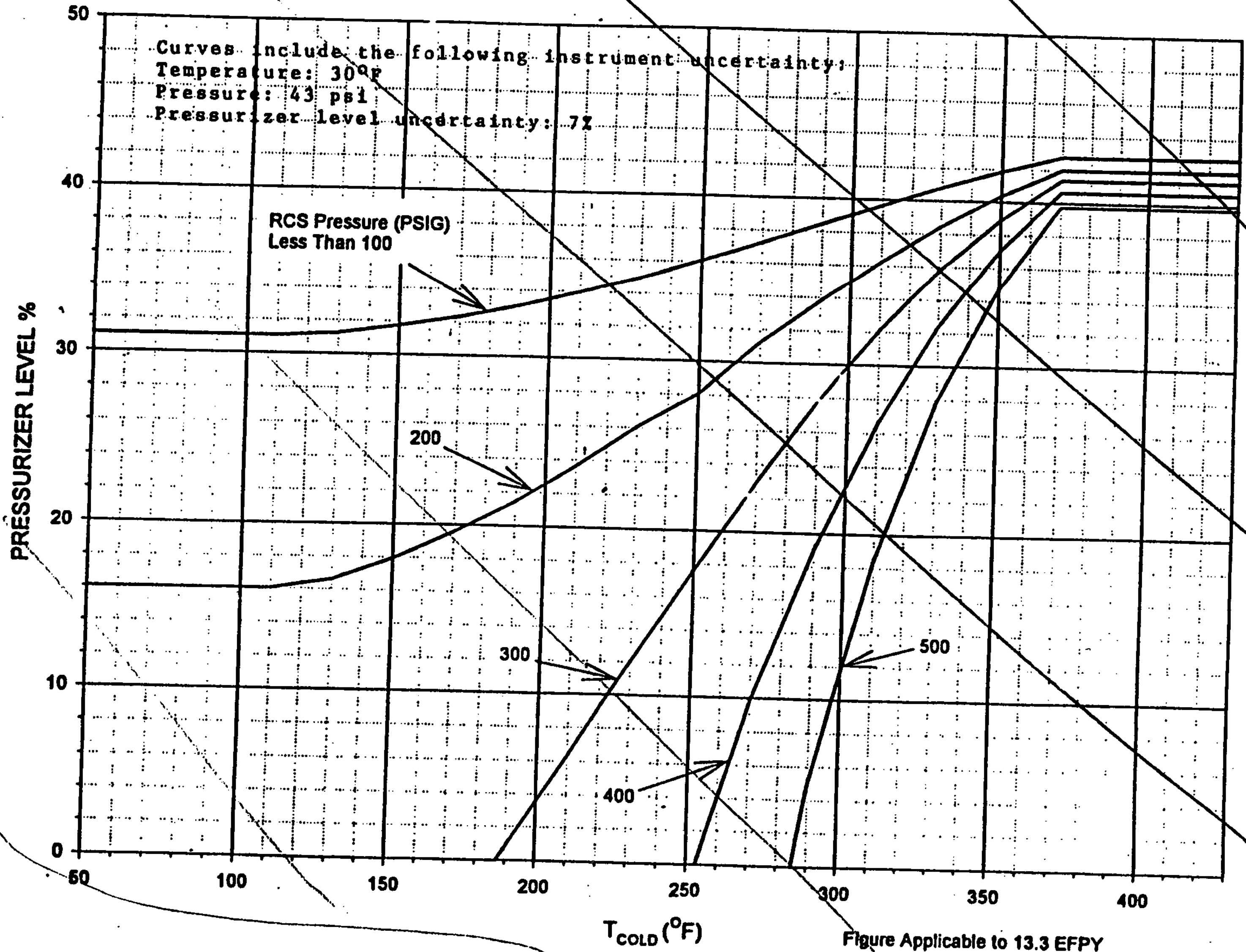
FIGURE 3.1.A-6

Pressurizer Limitations for OPS Inoperable (up to three charging pumps and/or one safety injection pump capable of feeding RCS), 13.3 EFPY




LA.1

ITS 3.4.12



Curves represent maximum allowable pressurizer level for the conditions defined.

Amendment No. 67, 101, 121, 179

SEE ITS 3.4.7 ITS 3.4.8

2) RCS temperature and the source range detectors are monitored hourly;

and

3) no operations are permitted which would reduce the boron concentration of the reactor coolant system.

LCO 3.4.12 and Applicability Note to LCO 3.4.12.a & b

~~8.~~ When the RCS average cold leg temperature ($T_{cold}$) is below 319°F, or when RHR is in service (i.e., not isolated from the RCS), no ~~safety injection~~ HHSI pumps, isolate accumulators, shall be ~~energized and aligned to feed~~ capable of injecting the RCS. (A.4) (A.11) (M.1)

LCO 3.4.12 Note 2 and 3

9. The requirements of 3.3.A.8 may be relaxed to allow one ~~safety injection~~ HHSI pump ~~energized and aligned~~ capable of injecting to feed the RCS under the following circumstances: (A.11)

LCO 3.4.12, Note 2 — a. emergency boration; OR

LCO 3.4.12, Note 3 — b. for pump testing, for a period not to exceed 8 hours; OR

LCO 3.4.12, Note 2 — c. loss of RHR cooling.

Reg. Act A.2 and A.3

10. The requirements of 3.3.A.8 may be further relaxed ~~when the RCS is < 200°F~~, such that two ~~safety injection~~ HHSI pumps may be energized and aligned to feed the RCS under the following circumstances: (A.7)

Reg. Act A.3.1 A.3.2

a. Verify immediately & 12 hours thereafter the RCS is vented with an opening greater than or equal to the size of one code pressurizer safety valve flange, OR (M.2)

Reg. Act A.2.1 A.2.2 A.2.3

b. Verify immediately & 12 hours thereafter ~~indicated~~ pressurizer level is at 0% and the plant is vented in accordance with Technical Specification 3.1.A.8.c.1. ~~(Alternate methods and instrumentation may be used to confirm actual RCS elevation.)~~ (M.2) (LA.3)

SEE ITS 3.6.6 ITS 3.6.7

B. Containment Cooling and Iodine Removal Systems

1. The reactor shall not be brought above the cold shutdown condition unless the following requirements are met:

a. The spray additive tank contains a minimum of 4000 gallons of solution with a sodium hydroxide concentration ≥35% and ≤38% by weight.

b. The five fan cooler-charcoal filter units and the two spray pumps, with their associated valves and piping, are operable.

2. The requirements of 3.3.B.1 may be modified to allow any one of the following components to be inoperable at one time:

TABLE 3.5-3 (Sheet 3 of 3)

INSTRUMENTATION OPERATING CONDITION FOR ENGINEERED SAFETY FEATURES

| | No. FUNCTIONAL UNIT | 1<br>NO. OF CHANNELS | 2<br>NO. OF CHANNELS TO TRIP | 3<br>MIN. NUMBER OF OPERABLE CHANNELS | 4<br>MIN. DEGREE OF REDUNDANCY | 5<br>OPERATOR ACTION IF CONDITIONS OF COL. 3 OR 4 CANNOT BE MET (Note 6) | |
|---|---|---|---|---|---|---|---|
| SEE ITS 3.3.5 | 4. LOSS OF POWER<br>a. 480v Bus Undervoltage Relay | 2/bus | 1/bus | 1/bus | 0 | See Note 1 | |
| | b. 480v Bus Degraded Voltage Relay | 2/bus | 2/bus | 2/bus (See Note 2) | 0 | See Note 1 | |
| LCO 3.4.12 | 5. ~~OVERPRESSURE PROTECTION SYSTEM (OPS)~~ | ~~3~~ | ~~2~~ | ~~2~~ | ~~1~~ | ~~See Note 7~~ | LA.2 |

SEE ITS 3.3.2 and ITS 3.3.5

Note 1. If the 138KV and 13.8KV sources of offsite power are available and the conditions of column 3 or 4 cannot be met within 72 hours, then the requirements of 3.7.C.1 or 2 shall be met.

Note 2. If one channel becomes inoperable, it is placed in the trip position and the minimum number of operable channels is reduced by one.

Note 3. Permissible to bypass if reactor coolant pressure is less than 2000 psig.

Note 4. Must actuate 2 switches simultaneously.

Note 5. The Minimum Number of Operable Channels and the Minimum Degree of Redundancy may be reduced to zero if the SI bypass is in the unblocked position.

Note 6. If the condition of Column 3 or 4 cannot be met, the reactor shall be placed in the hot shutdown condition, utilizing normal operating procedures, within 4 hours of the occurrence. If the conditions are not met within 24 hours of the occurrence, the reactor shall be placed in the cold shutdown condition, or the alternate condition, if applicable, within an additional 24 hours.

LCO 3.4.12 Note 7. ~~Refer to Specification 3.1.A.8.~~ LA.2

SEE ITS 3.3.2 Note 8. Main steam isolation valves may be closed in lieu of going to cold shutdown if the circuitry associated with closing the valves is the only portion inoperable.

Amendment No. ~~38~~, ~~44~~, ~~54~~, ~~87~~, ~~113~~, 151

TSCR 96-124 not shown.
No impact for ITS 3.4.12

Table 3.5-3 (Sheet 3 of 3)

INSTRUMENTATION OPERATING CONDITION FOR ENGINEERED SAFETY FEATURES

| No. FUNCTIONAL UNIT | 1 NO. OF CHANNELS | 2 NO. OF CHANNELS TO TRIP | 3 MIN. NUMBER OF OPERABLE CHANNELS | 4 MIN. DEGREE OF REDUNDANCY | 5 OPERATOR ACTION IF CONDITIONS OF COL. 3 OR 4 CANNOT BE MET (Note 6) |
|---|---|---|---|---|---|
| 4. LOSS OF POWER a. 480v Bus Undervoltage Relay | 2/bus | 1/bus | 1/bus | 0 | See Note 1 |
| b. 480v Bus Degraded Voltage Relay | 2/bus | 2/bus | 2/bus (See Note 2) | 0 | See Note 1 |
| 5. OVERPRESSURE PROTECTION SYSTEM (OPS) | 3 | 2 | 2 | 1 | See Note 7 |
| 6. ENGINEERED SAFETY FEATURE RELAY LOGIC (TRAIN) | 2 | 1 | 2 | 1 | Cold Shutdown |

Note 1. If the 138KV and 13.8 KV sources of offsite power are available and the conditions of column 3 or 4 cannot be met within 72 hours, then the requirements of 3.7.C.1 or 2 shall be met. Implementation of this note supersedes Note 6.

Note 2. If one channel becomes inoperable, it is placed in the trip position and the minimum number of operable channels is reduced by one.

Note 3. Permissible to bypass if reactor coolant pressure is less than 2000 psig.

Note 4. Must actuate 2 switches simultaneously.

Note 5. The Minimum Number of Operable Channels and the Minimum Degree of Redundancy may be reduced to zero if the SI bypass is in the unblocked position.

Note 6. If the conditions specified in Columns 3 or 4 cannot be met within 6 hours, then the reactor shall be placed in the hot shutdown condition, utilizing normal operating procedures, within the following 6 hours. If the conditions of Columns 3 and 4 are not met within 24 hours of the occurrence, the reactor shall be placed in the cold shutdown condition, or the alternate condition, if applicable, within an additional 24 hours.

The above actions are modified by Specification 3.5.4 to allow testing for up to 8 hours.

Note 7. Refer to Specification 3.1.A.8. Implementation of this note supersedes Note 6.

Note 8. Main steam isolation valves may be closed in lieu of going to cold shutdown if the circuitry associated with closing the valves is the only portion inoperable.

ITS 3.4.12

A.10

add Note to SR 3.4.12.6 — L.1

**TABLE 4.1-1** (Sheet 5 of 6)

| | Channel Description | Check | Calibrate | Test | Remarks |
|---|---|---|---|---|---|
| SEE ITS 3.3.3 | 37. Core Exit Thermocouples | D | N.A. | 18M | |
| SR 3.4.12.4 SR 3.4.12.6 | 38. Overpressure Protection System (OPS) | D 24 hours SR 3.4.12.4 & Note | 18M (1) SR 3.4.12.7a | 24M SR 3.4.12.6 | 1) Calibration frequency for OPS sensors (RCS pressure and temperature) is 24 months. SR 3.4.12.7.b |
| See ITS Master Markup | 39. Reactor Trip Breakers | N.A. | N.A. | TM(1) | 1) Independent operation of under-voltage and shunt trip attachments |
| | | | | 24M(2) | 2) Independent operation of under-voltage and shunt trip from Control Room manual push-button |
| | 40. Reactor Trip Bypass Breakers | N.A. | N.A. | (1) | 1) Manual shunt trip prior to each use |
| | | | | 24M(2) | 2) Independent operation of under-voltage and shunt trip from Control Room manual push-button |
| | | | | 24M(3) | 3) Automatic undervoltage trip |
| | 41. Reactor Vessel Level Indication System (RVLIS) | D | 18M***** | N.A. | |
| | 42. Ambient Temperature Sensors Within the Containment Building | D | 24M | N.A. | |
| | 43. River Water Temperature # (installed) | S | 18M | N.A. | |
| | 44. River Water Temperature # (portable) | S (1) | Q (2) | N.A. | 1) Check against installed instrumentation or another portable device. 2) Calibrate within 30 days prior to use and quarterly thereafter. |
| | 45. Steam Line Flow | S | 24M | Q | Engineered Safety Features circuits only |



TSCR 98-043 not shown. No impact on ITS 3.4.12

ITS 3.4.12

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.4.12:
# "Low Temperature Overpressure Protection (LTOP)"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.4.12 - Low Temperature Overpressure Protection (LTOP)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the Improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.1.A.8 specifies requirements for the Overpressure Protection System (OPS) or an RCS vent when the RCS temperature is at or below 319°F; CTS 3.1.A.1.h specifies limitations for Reactor Coolant Pump (RCP) starting when RCS cold leg temperature (Tcold) is at or below 319°F; and CTS 3.3.A.8 establishes requirements for limiting high head coolant injection capability when Tcold is at or below 319°F.

Additionally, CTS 3.3.A.8 establishes requirements for limiting high head coolant injection capability when RHR is in service (i.e., not isolated from the RCS) to protect the RHR system from overpressurization. ITS LCO 3.4.12 organizes and maintains these requirements, including protection of the RHR system, as a Limiting Condition for Operation that provides low temperature overpressure protection (LTOP). This reorganization of requirements is an administrative change with no impact on safety because there is no change to existing requirements except those identified and justified elsewhere in the discussion of changes.

A.4 CTS 3.1.A.8 specifies requirements for the Overpressure Protection System (OPS) or an RCS vent when the RCS temperature is at or below 319°F; CTS 3.1.A.1.h specifies limitations for Reactor Coolant Pump (RCP) starting when RCS cold leg temperature (Tcold) is below 319°F; and CTS 3.3.A.8 establishes requirements for limiting high head coolant injection capability when Tcold is at or below 319°F. Additionally, CTS 3.3.A.8 establishes requirements for limiting high head coolant injection capability when RHR is in service (i.e., not isolated from the RCS) to protect the RHR system from over-pressurization.

ITS LCO 3.4.12 establishes the Applicability for LTOP as Mode 4 when the average of RCS cold leg temperatures is < 319°F and in Modes 5 and 6 when the reactor vessel head is on. Additionally, ITS LCO 3.4.12 is Applicable whenever the RHR System is not isolated from the RCS except that a Note to LCO 3.4.12.a and b excludes the requirements for OPS or an RCS vent when the average RCS cold leg temperature is ≥ 319°F.

The difference between ITS and CTS is that the ITS uses the more definitive statement that LTOP is required when the average of RCS cold leg temperatures is < 319°F. This change is needed so that the ITS LCO 3.4.12 Applicability is consistent with ITS LCO 3.4.10, Pressurizer Safety Valves, which requires that pressurizer safety valves provide overpressure protection when "the average of the" RCS cold leg temperatures is ≥ 319°F. Additionally, the ITS clarifies the CTS applicability with an explicit recognition that LTOP is not required when the reactor vessel head is removed. These are administrative changes with no significant adverse impact on safety because the change is a more explicit statement of a reasonable interpretation of existing requirements.

A.5 CTS 3.1.A.1.h specifies reactor coolant pump (RCP) starting prerequisites when an RCP is "started or jogged" if RCS temperature is ≤ 319°F. ITS LCO 3.4.12 maintains these RCP starting prerequisites as ITS SR 3.4.12.8 and ITS SR 3.4.12.9 that are required before starting any RCP if RCS average temperature is < 319°F. The ITS SR 3.4.12.8 and ITS SR 3.4.12.9 include the clarification that these requirements are also applicable when an RCP is jogged. This is an administrative change with no adverse impact on safety because there is no change to the existing requirement to verify RCP starting prerequisites when an RCP is jogged.

A.6 CTS 3.1.A.8.a.1 uses the term "armed" in conjunction with the term operable to establish the required status of the Overpressure Protection System (OPS). The CTS Bases explain that "arming" means that the motor operated block valve associated with each PORV is in the open position. The OPS is "armed" if the block valve is either opened automatically by the OPS when the RCS average temperature is less than 319°F or opened manually by the control room operator.

ITS 3.4.12 and ITS SR 3.4.12.5 maintain this requirement that the motor operated block valve associated with each PORV must be open for OPS to be Operable; however, the term "armed" is not used to describe this condition. ITS SR 3.4.12.5 requires periodic verification that each block valve is open as a condition of Operability of the OPS. Therefore, the term "armed" can be deleted because the status of the block valve is included within the definition of Operability of the OPS. This is an administrative change with no impact on safety.

A.7 CTS 3.3.A.10 provides an exception to CTS 3.3.A.8 and allows up to two HHSI pumps to be aligned to the RCS and energized if restrictions on RCS vent size and/or pressurizer level are met. However, CTS 3.3.A.10

stipulates that this allowance only applies when RCS average temperature is < 200°F. ITS LCO 3.4.12, Required Actions A.2.1, A.2.2 and A.2.3, maintain the allowances in CTS 3.3.A.10 and allow up to two HHSI pumps to be aligned to the RCS and energized if restrictions on RCS vent size and pressurizer level are met. However, the restriction that this allowance only applies when RCS average temperature is < 200°F is deleted. This change is acceptable because the restriction that RCS average temperature is < 200°F is a necessary condition for maintaining the RCS vented. This is an administrative change with no adverse impact on safety because there is no significant change to the existing requirement.

A.8 CTS 3.1.A.8.d requires that if the PORV's or the RCS vent(s) are used to mitigate an RCS pressure transient, then a special report be submitted to the NRC within 30 days pursuant to Specification 6.9.2.j. CTS 6.9.2 is a list of the special reports that must be submitted to the NRC for a variety of activities or events identified in the CTS.

ITS 3.4.12 does not include an explicit requirement for the submittal of a special report if the PORV's or the RCS vent(s) are used to mitigate an RCS pressure transient. This change is acceptable because requirements with respect to reportable events are included in 10 CFR 50.72 and 10 CFR 50.73, and need not be repeated in the ITS. This is an administrative change with no adverse impact on safety.

A.9 CTS 3.1.A.8.c includes several options for Actions for failure to meet LTOP requirements because of a loss of redundancy or loss of function for the Overpressure Protection System (OPS); however, no Actions are provided if LTOP requirements are not met for other reasons.

ITS LCO 3.4.12, Condition F and Required Action F.1, require that the RCS must be depressurized and a vent path ≥ 2.00 square inches must be established within 8 hours when low temperature protection requirements are not met for reasons other than those identified as LCO 3.4.12 Conditions. This is an administrative change with no adverse impact on safety because depressurizing the RCS and establishing a vent path of greater than 2.00 square inches satisfies both CTS and ITS requirements for LTOP. Therefore, this is an explicit statement of an existing requirement.

A.10 CTS Table 4.1-1, Item 38, includes a requirement for a daily channel check of the Overpressure Protection System (OPS). ITS SR 3.4.12.4 maintains the requirement to perform a Channel Check of the OPS instrument channels every 24 hours; however, ITS SR 3.4.12.4 is modified by a note that includes an explicit statement that Channel Checks of OPS instrument channels are only required if the OPS is being used to satisfy LTOP requirements. This change is needed because ITS LCO 3.4.12 allows low temperature overpressure protection to be established using the Overpressure Protection System (OPS), RCS venting, or, if OPS or the associated PORVs are inoperable, limits on pressurizer level. This is an administrative change with no adverse impact of safety because it is an explicit statement of a reasonable interpretation of the existing requirement to perform OPS channel checks only when OPS is required to be Operable.

A.11 CTS 3.3.A.8 specifies that no safety injection (HHSI) pump shall be "energized and aligned" to feed the RCS when LTOP is required. ITS LCO 3.4.12 specifies that no safety injection (HHSI) pump shall be "capable of injecting into the RCS" when LTOP is required. This is an administrative change with no adverse impact on safety because there is no change to the existing requirement.

MORE RESTRICTIVE

M.1 CTS 3.1.A.8 does not include any requirements to isolate or depressurize the ECCS accumulators to limit injection capability when low temperature overpressure protection is required. CTS 3.3.A.8 does not include any requirements to isolate or depressurize the ECCS accumulators to limit injection capability when the RHR system is aligned to the RCS. However, current operating procedures do require that safety injection accumulators are isolated when RCS pressure is reduced below 1000 psig to prevent an unplanned injection.

ITS LCO 3.4.12 includes a requirement that ECCS accumulators are isolated (discharge isolation valves closed and de-energized) when LTOP is required to be Operable. Additionally, ITS LCO 3.4.12, Note 1, is added to clarify that accumulator isolation is required only if the accumulator is not depressurized to less than the maximum RCS pressure allowed by the P/T limit curves provided in the PTLR.

In conjunction with this change, ITS LCO 3.4.12, Condition B and associated Required Action B.1, are added to require that an unisolated ECCS accumulator must be isolated within 1 hour. (Note that this Action applies to an isolation valve that is closed but not de-energized.) Additionally, ITS LCO 3.4.12, Condition C and associated Required Actions, are added to require that if the accumulator cannot be isolated and de-energized within one hour, then the accumulator must be depressurized within 12 hours or the RCS average temperature increased to exit ITS LCO 3.4.12 Applicability and the RHR System isolated within 12 hours. In conjunction with new requirement for accumulator isolation, ITS SR 3.4.12.2 is added to require verification every 12 hours that ECCS accumulators are isolated or depressurized when LTOP is required.

These changes are needed because the ECCS accumulators contain significant amounts of stored energy and have the potential to cause unacceptable pressure excursions if inadvertently injected even if other LTOP requirements are satisfied. Inadvertent injection of a charged accumulator into the RCS is not assumed in LTOP capability. This change is acceptable because it does not introduce any operation which is un-analyzed while establishing the following: explicit requirements that minimize the potential for inadvertent ECCS accumulator injection when LTOP is required or the RHR system is not isolated from the RCS; requirements for periodic verification that ECCS accumulators are isolated or depressurized; and, Required Actions if a requirement is not met. This change has no significant adverse impact on safety.

M.2 CTS 3.3.A.8 specifies that no safety injection (HHSI) pump shall be energized and aligned to feed the RCS when LTOP is required. CTS 3.3.A.10 provides an exception to this requirement and allows up to two HHSI pumps to be aligned to the RCS and energized if restrictions on RCS vent size and/or pressurizer level are met. However, CTS 3.3.A.8 and CTS 3.3.A.10 do not require periodic verification that restrictions on safety injection pump or the alternate restrictions on RCS vent size and/or pressurizer level are met. Additionally, CTS 3.3.A.8 and CTS 3.3.A.10 do not specify any required actions if these requirements are not met. CTS 3.0, a requirement for plant shutdown and cooldown, is not appropriate because plant cooldown would exacerbate the condition.

ITS SR 3.4.12.1 is added to require verification every 12 hours that no HHSI pumps are capable of injecting into the RCS. If ITS SR 3.4.12.1 is not met, ITS LCO 3.4.12, Condition A and associated Required Actions apply. ITS LCO 3.4.12, Required Action A.1, requires immediate action to eliminate capability for HHSI injecting into the RCS. Alternately, ITS LCO 3.4.12, Required Actions A.2 and A.3, maintain the allowances in CTS 3.3.A.10 and allow up to two HHSI pumps to be aligned to the RCS and energized if restrictions on RCS vent size and/or pressurizer level are met. ITS LCO 3.4.12, Required Actions A.2 and A.3, differ from CTS 3.3.A.10 in that verification every 12 hours that requirements are met is required.

These changes are needed to require periodic verification that LTOP analysis assumptions regarding limits on high head injection capacity are met and to provide explicit actions if these requirements are not met. These changes are acceptable because they do not introduce any operation which is un-analyzed while requiring specific required actions if a requirement is not met and periodic surveillances to identify when requirements are not met. Therefore, this change has no adverse impact on safety.

M.3 CTS 3.1.A.1.h.3 establishes RCP starting prerequisites when RCS temperature is < 319°F when OPS is inoperable. However, CTS 3.1.A.1.h.3 includes an allowance that the special restrictions on pressurizer level for RCP pump starts with OPS inoperable are not mandatory if the RCS average temperature has been < 319°F for less than 8 hours.

ITS LCO 3.4.12 maintains these RCP starting prerequisites as ITS SR 3.4.12.8 and ITS SR 3.4.12.9: however, the relaxation of requirements for additional restrictions on pressurizer level for the first 8 hours that RCS average temperature is < 319°F is not provided in the ITS. This change is needed because the CTS 3.1.A.1.h.1 allowance for 8 hours increases the potential for an unacceptable pressure transient during an RCP pump start. This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while providing a higher degree of assurance that an unacceptable pressure transient will be avoided during an RCP start. Therefore, this change has no adverse impact on safety.

M.4 CTS 3.1.A.1.h specifies RCP starting prerequisites that must be met if an RCP is started when RCS temperature ≤ 319°F.

ITS LCO 3.4.12 maintains these RCP starting prerequisites as ITS SR 3.4.12.8 and ITS SR 3.4.12.9. Theses SRs have a required Frequency of "Within 15 minutes prior to starting any RCP" which means that these prerequisites must be performed within 15 minutes prior to pump start and must be met at the time of the pump start. This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while providing better administrative assurance that RCP starting prerequisites will be met at the time RCPs are started. Therefore, this change has no adverse impact on safety.

M.5 CTS 3.1.A.8.c.3 requires that pressurizer level be restricted in accordance with 3.1.A-5 and 3.1.A-6 (See ITS 3.4.12, DOC LA.1) within 8 hours if requirements for OPS and/or RCS venting are not met. Under the same conditions, ITS LCO 3.4.12, Required Action E.3, maintains this requirement (i.e., verify pressurizer level, RCS pressure, and RCS injection capability are within limits specified in PTLR for OPS not Operable within 8 hours). However, ITS LCO 3.4.12, Required Action E.3, requires re-verification every 12 hours that this restrictions are being met.

This change is needed to require periodic verification that LTOP analysis assumptions regarding limits on pressurizer level, RCS pressure, and RCS injection capability are being met. This change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification that requirements are met. Therefore, this change has no adverse impact on safety.

M.6 CTS 3.1.A.8 includes the option of using an RCS vent with an equivalent opening of at least 2.00 square inches to provide LTOP when the RCS temperature is at or below 319°F; however, CTS 3.1.A.8 does not require periodic verification that requirements are met if this option is used.

ITS LCO 3.4.12.b maintains the option of using an RCS vent to provide LTOP and ITS SR 3.4.12.3 is added to require periodic verification that requirements are satisfied when complying with ITS LCO 3.4.12.b. The required Frequency for this periodic verification is every 12 hours if the vent path is established using unlocked valves and every 31 days if the vent path is established using locked valves.

The addition of ITS SR 3.4.12.3 is needed to require periodic verification that LTOP analysis assumptions regarding RCS vent status and capacity are met. The required Frequency is based on engineering judgment and recognizes that procedural controls governing locked and unlocked valve operation provide a high degree of assurance that the vent path can be maintained. This change is acceptable because they do not introduce any operation which is un-analyzed while requiring specific required actions if a requirement is not met and periodic surveillances to identify when requirements are not met. Therefore, this change has no adverse impact on safety.

M.7 CTS 3.1.A.8.a.1 allows using the Overpressure Protection System (OPS) to provide LTOP when the RCS temperature is at or below 319°F if the OPS is "armed" (See ITS 3.4.12, DOC A.6) and operable; however, CTS 3.1.A.8.a does not require periodic verification that the OPS is "armed" if this option is used. CTS 3.1.A.8.a uses the term "armed" to mean that the motor operated block valve associated with each PORV is in the open position. The OPS is "armed" if the block valve is either opened automatically by the OPS or opened manually by control room operators.

ITS SR 3.4.12.5 is added to require verification every 72 hours that the block valve is open for each required PORV. The addition of ITS SR 3.4.12.5 is needed as part of the periodic verification that the Overpressure Protection System is Operable. The 72-hour Frequency is considered adequate to ensure that PORV block valve remains open because the PORV block valves are opened automatically by the OPS when below the OPS arming temperature if the valve control is positioned to auto and because valve position indication is available to the operator in the control room. This change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic surveillances to identify when requirements are not met. Therefore, this change has no adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS Table 4.1-1, Item 38, requires a test of the Overpressure Protection System every 24 months. ITS SR 3.4.12.6 maintains this requirement by requiring a Channel Operational Test on each required PORV (excluding PORV actuation) every 24 months; however, the Frequency includes an allowance that the initial performance is not required until 12 hours after decreasing RCS average temperature to < 319°F.

This change, an allowance that the initial performance is not required until 12 hours after decreasing RCS average temperature to < 319°F, is needed because the COT cannot be performed until plant conditions exist that allow the PORV lift setpoint to be reduced to the LTOP setting. This allowance is acceptable because of the high probability that the test will demonstrate that the OPS is Operable and the low probability of an overpressure event during the 12-hour period. Additionally, RCS average temperature will typically remains in the upper part of the LTOP range and a bubble is typically maintained in the pressurizer during this period. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

LA.1 CTS 3.1.A.8 and CTS 3.1.A.1.h establish RCS pressure and temperature limits for low temperature overpressure protection when RCS temperature is < 319°F and the RCP starting prerequisites when RCS temperature is < 319°F. These pressure and temperature limits are established by reference to the following CTS figures:

| | |
|---|---|
| Figure 3.1.A-1: | Secondary Side Limitations for RCP Start with Secondary Side Hotter than Primary, 13.3 EFPY; |
| Figure 3.1.A-2: | Maximum Allowable Nominal PORV Setpoint for the Low Temperature Overpressure Protection System, 13.3 EFPY; |
| Figure 3.1.A-5: | Pressurizer Limitations for OPS Inoperable (Up to One Charging Pump Capable of feeding the RCS, 13.3 EFPY; |
| Figure 3.1.A-6: | Pressurizer Limitations for OPS Inoperable (Up to Three Charging Pumps and/or One Safety Injection Pump Capable of feeding the RCS, 13.3 EFPY; |

These Figures are not retained in the ITS and are relocated to the Pressure Temperature Limits Report (PTLR) in accordance with the guidance provided in Generic Letter 96-03. This change is needed because the pressure temperature limits being moved to the PTLR are revised on a periodic basis which currently requires Technical Specification changes.

This change is acceptable because ITS LCO 3.4.12 maintains the requirement to meet these pressure and temperature limits when low temperature overpressure protection is required and ITS 5.6.6, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR), maintains detailed requirements for the establishment of these pressure and temperature limits including a requirement to use analytical methods previously reviewed and approved by the NRC and a requirement to provide a copy of the PTLR to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight is maintained for the information being relocated out of the Technical Specifications.

LA.2 CTS Table 3.5-3, Item 5, establishes requirements for the 3 channels of RCS pressure and 3 channels of RCS temperature needed to support of Operability of the Overpressure Protection System (OPS). For both the pressure and temperature function 2 out of 3 channels are required for actuation; therefore, CTS Table 3.5-3, Item 5, requires 2 operable channels and a minimum degree of redundancy of 1 (i.e., operation may continue indefinitely with 1 of the 3 channels in trip).

ITS 3.4.12.a requires the Operability of the Overpressure Protection System; however, requirements for the RCS pressure and temperature channels are relocated to the LCO Section of the ITS 3.4.12 Bases. The ITS 3.4.12 Bases specify that the OPS is OPERABLE for LTOP when there are three OPERABLE RCS pressure channels and three OPERABLE RCS temperature channels. The OPS is still OPERABLE when an inoperable RCS pressure or temperature channel is in the tripped condition.

This change, which allows the description of the design of the of the OPS RCS pressure and temperature logic to be maintained in the FSAR and the detailed description of the requirements for Operability of these instrument channels to be maintained in the ITS Bases, is consistent with the approach used in NUREG-1431 for all Limiting Conditions for Operation (LCOs). This approach is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, and ITS 5.5.13, Technical Specifications (TS) Bases Control Program, are designed to assure that changes to the FSAR and ITS Bases do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 and ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of FSAR and Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

LA.3 CTS 3.3.A.10 allows up to two HHSI pumps to be capable of injecting into the RCS when RCS temperature is < 319°F and/or RHR is not isolated if restrictions on RCS vent size and pressurizer level are met. The restriction on pressurizer level requires that "indicated" pressurizer level is at 0% with an allowance that "alternate methods and instrumentation may be used to confirm actual RCS elevation."

ITS LCO 3.4.12, Required Actions A.2.1, A.2.2 and A.2.3, maintain the allowances in CTS 3.3.A.10 and allow up to two HHSI pumps to be aligned to the RCS and energized if restrictions on RCS vent size and pressurizer level are met. However, the information that "indicated" pressurizer level may be used and the allowance that "alternate methods and instrumentation may be used to confirm actual RCS elevation" are relocated to the ITS LCO 3.4.12 Bases.

This change is acceptable because ITS LCO 3.4.12, Required Action A.2.2, maintains the requirement to have pressurizer level ≤ 0% as a condition for having HHSI pumps capable of injecting into the RCS. The information about how pressurizer level can be verified to meet this requirement is a design issue that is more appropriately controlled in the ITS Bases. Maintaining this information in the Bases is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, and ITS 5.5.13, Technical Specifications (TS) Bases Control Program, are designed to assure that changes to the ITS Bases do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated to the Technical Specification Bases.

LA.4 CTS 3.1.A.8 specifies requirements for the Overpressure Protection System (OPS) or an RCS vent when the RCS temperature is < 319°F. However, if these requirements are not met, CTS 3.1.A.8.c.2 requires that RCS temperature is increased to > 411°F (i.e., a temperature significantly higher than the LTOP applicability). Note that 319°F is the LTOP arming temperature and 411°F is the minimum temperature at which the pressurizer safety valves lift prior to violation of the 10 CFR 50, Appendix G, limits.

Under the same conditions (LTOP requirements not met), ITS LCO 3.4.12, Required Action E.2, requires that RCS temperature is increased to ≥ 319°F which places the plant outside the LTOP Applicability. The requirement that RCS temperature is increased to > 411°F is relocated to the Technical Requirements Manual (TRM) and is included with the administrative controls governing operation above the LTOP arming temperature but below the minimum temperature at which the pressurizer safety valves lift prior to violation of the 10 CFR 50, Appendix G, limits.

This change is needed because ITS 3.0.2 specifies that Completing the Required Actions is not required when the plant is placed outside the LCO Applicability. This change is acceptable because establishing administrative controls governing operation above the LTOP arming temperature but below the minimum temperature at which the pressurizer safety valves lift prior to violation of the 10 CFR 50, Appendix G, limits is consistent with the guidance provided in Generic Letter 88-011, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations. GL 88-011 states that automatic, or passive, protection of the P-T limits will not be required but administratively controlled when in the upper end of the temperature range. ITS LCO 3.4.12 maintains the requirements for automatic protection of the Appendix G P-T limits below the CTS LTOP arming temperature.

The Quality Assurance Plan will be revised to specify that requirements in the TRM are part of the facility as described in the FSAR and that changes to the TRM can be made only in accordance with the requirements of 10 CFR 50.59. Therefore, this change is acceptable because there is no change to the existing requirements by the relocation of requirements to the TRM and future changes to the TRM will be controlled in accordance with 10 CFR 50.59.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

# Technical Specification 3.4.12:
# "Low Temperature Overpressure Protection (LTOP)"

## PART 4:

## No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.4.12 - Low Temperature Overpressure Protection (LTOP)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   CTS Table 4.1-1, Item 38, requires a test of the Overpressure Protection System every 24 months. ITS SR 3.4.12.6 maintains this requirement by requiring a Channel Operational Test on each required PORV (excluding PORV actuation) every 24 months; however, the Frequency includes an allowance that the initial performance is not required until 12 hours after decreasing RCS average temperature to < 319°F. This change, an allowance that the initial performance is not required until 12 hours after decreasing RCS average temperature to < 319°F, is needed because the COT cannot be performed until plant conditions exist that allow the PORV lift setpoint to be reduced to the LTOP setting.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the OPS Operability status is not related to the precursor of any analyzed accident. This change will not result in a significant increase in the consequences of an accident previously evaluated because of the high probability that the test will demonstrate that the OPS is Operable and the low probability of an overpressure event during the 12-hour period. Additionally, RCS average temperature will typically remains in the upper part of the LTOP range and a bubble is typically maintained in the pressurizer during this period.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because of the high probability that the test will demonstrate that the OPS is Operable and the low probability of an overpressure event during the 12-hour period. Additionally, RCS average temperature will typically remains in the upper part of the LTOP range and a bubble is typically maintained in the pressurizer during this period.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.4.12:
# "Low Temperature Overpressure Protection (LTOP)"

## PART 5:

## NUREG-1431
## Annotated to show differences between
## NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.4.12

This ITS Specification is based on NUREG-1431 Specification No. 3.4.12
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| WOG-051 R1 | | CHARGING PUMP SWAP LTOP ALLOWANCE | Rejected by TSTF | Not Incorporated | N/A |
| WOG-051 R2 | | CHARGING PUMP SWAP LTOP ALLOWANCE | Rejected by TSTF | Not Incorporated | N/A |
| WOG-067 R1 | | RELOCATE LTOP ARMING TEMPERATURE TO PTLR | Rejected by TSTF | Not Incorporated | N/A |
| WOG-100 | | EXEMPT SRS ON LTOP EQUIPMENT NOT USED TO SATISFY THE LCO | TSTF Review | Not Incorporated | N/A |
| WOG-104 | | LTOP VENT PATH SR FREQUENCY | TSTF Review | Not Incorporated | N/A |
| WOG-111 | | CORRECT APPLICABILITY FOR LTOP SPECIFICATIONS | TSTF Review | Not Incorporated | N/A |

## 3.4 REACTOR COOLANT SYSTEM (RCS)

<3.1.A.8>
<3.3.A.8>

### 3.4.12 Low Temperature Overpressure Protection (LTOP) System

<3.1.A.8>
<3.3.A.8>
<DOC M.1>
<DOC A.11>

CLB.1

LCO 3.4.12 ~~An~~ LTOP ~~System~~ shall be OPERABLE with ~~a maximum of one high pressure injection (HPI) pump~~ ~~[and one charging pump]~~ capable of injecting into the RCS and the accumulators isolated and either ~~a or b below~~ of the following:

Insert: 3.4-27-01

Insert: 3.4-27-02

Insert: 3.4-27-03

a. Two RCS relief valves, ~~as follows:~~

1. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

[2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ [436.5] psig and ≤ [463.5] psig, or]

[3. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint ≥ [436.5] psig and ≤ [463.5] psig].

b. The RCS depressurized and an RCS vent of ≥ [2.07] square inches.

CLB.1

Insert: 3.4-27-04

<3.1.A.8>
<3.3.A.8>
<DOC A.4>

APPLICABILITY: MODE 4 when ~~all~~ average RCS cold leg temperature is ≤ S [275] 319 < °F,
MODE 5,
MODE 6 when the reactor vessel head is on.

DB.1

<DOC M.1>

1.

-------------------------NOTE S-------------------------
Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.
----------------------------------------------------------

Insert: 3.4-27-05

**INSERT: 3.4-27-01**

<3.3.A.8> no high head safety injection (HHSI) pumps

**INSERT: 3.4-27-02**

<DOC M.1> discharge isolation valves closed and de-energized

**INSERT: 3.4-27-03**

<3.3.A.8>
<DOC M.4>

---------------------------Note---------------------------

LCO 3.4.12.a and LCO 3.4.12.b are not Applicable when average RCS cold leg temperature is ≥ 319°F.

----------------------------------------------------------

<3.1.A.8.a.1>
<DOC LA.1>
<DOC A.6>

a. The Overpressure Protection System (OPS) OPERABLE with two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR;

<u>OR</u>

<3.1.A.8.a.2> b. The RCS depressurized with an RCS vent of ≥ 2.00 square inches.

**INSERT: 3.4-27-04**

<3.3.A.8> Whenever the RHR System is not isolated from the RCS.

**INSERT: 3.4-27-05**

<3.3.A.9.a>
<3.3.A.9.c>

2. One HHSI pump may be made capable of injecting into the RCS as needed to support emergency boration or to respond to a loss of RHR cooling.

<3.3.A.9.b>

3. One HHSI pump may be made capable of injecting into the RCS for a period not to exceed 8 hours to perform pump testing.

Insert:
3.4-28-01

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| ~~A. Two or more [HPI] pumps capable of injecting into the RCS.~~ | ~~A.1~~ | ~~Initiate action to verify a maximum of [one] [HPI] pump is capable of injecting into the RCS.~~ | ~~Immediately~~ |
| ~~B. Two or more charging pumps capable of injecting into the RCS.~~ | ~~B.1~~ | ~~---------NOTE--------- Two charging pumps may be capable of injecting into the RCS during pump swap operation for ≤ 15 minutes. ---------------------~~ | |
| | | ~~Initiate action to verify a maximum of [one] charging pump is capable of injecting into the RCS.~~ | ~~Immediately~~ |
| ~~C.~~ B. An accumulator ~~not isolated~~ when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR. | ~~C.1~~ B.1 | ~~Isolate affected accumulator.~~ | 1 hour |

<DOC M.1>

Insert:
3.4-28-02

Insert:
3.4-28-03

(continued)

INSERT: 3.4-28-01 (CLB.1)

| | | | |
|---|---|---|---|
| <3.3.A.8> <DOC M.2> | A. One or more HHSI pump(s) capable of injecting into the RCS . | A.1 Initiate action to verify no HHSI pumps are capable of injecting into the RCS. | Immediately |
| | | OR | |
| <3.3.A.10> <3.3.A.10.b> | | A.2.1 Verify RCS is vented with opening ≥ 2.00 square inches.<br><br>AND | Immediately |
| <3.3.A.10.b> <DOC LA.3><br><br><DOC M.2> | | A.2.2 Verify pressurizer level is ≤ 0%.<br><br>AND | Immediately<br><br>AND<br><br>Once per 12 hours |
| <3.3.A.10><br><br><DOC M.2> | | A.2.3 Verify no more than two HHSI pumps are capable of injecting into the RCS. | Immediately<br><br>AND<br><br>Once per 12 hours |
| | | OR | |
| <3.3.A.10.a> | | A.3.1 Verify RCS is vented with opening greater than or equal to one code pressurizer safety valve flange.<br><br>AND | Immediately |
| <3.3.A.10><br><br><DOC M.2> | | A.3.2 Verify no more than two HHSI pumps are capable of injecting into the RCS | Immediately<br><br>AND<br><br>Once per 12 hours |

**INSERT: 3.4-28-02**

discharge isolation valve not closed and de-energized

**INSERT: 3.4-28-03**

Close and de-energize isolation valve for affected accumulator.

ACTIONS (continued)

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <DOC M.1> | ~~D~~. C Required Action and associated Completion Time of Condition ~~C~~ B not met. Insert: 3.4-29-01 | ~~D~~.1 C.1 Increase average RCS cold leg temperature to ~~>~~ ≥ ~~[275]~~ 319 °F. | 12 hours |
| | | OR | |
| <DOC M.1> | | ~~D~~.2 C.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR. | 12 hours |
| <3.1.A.8.b> | ~~E~~. D One required ~~RCS relief valve~~ PORV inoperable ~~in MODE 4~~. | ~~E~~.1 Restore required ~~RCS relief valve~~ PORV to OPERABLE status. | 7 days |
| CLB.1 | ~~F. One required RCS relief valve inoperable in MODE 5 or 6.~~ | ~~F.1 Restore required RCS relief valve to OPERABLE status.~~ | ~~24 hours~~ |

(continued)

**INSERT: 3.4-29-01**

| | | |
|---|---|---|
| <DOC M.1> | <u>AND</u><br>C.1.2 Isolate the RHR System from the RCS. | 12 hours |

ACTIONS (continued)

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <3.1.A.8.c> <3.1.A.8.c.1> | ~~G~~. E Two required ~~RCS relief valves~~ PORVs inoperable. OR Required Action and associated Completion Time of Condition ~~A, [B,] B, E, or F~~ C or D not met. | ~~G~~.1 E Depressurize RCS and establish RCS vent of ≥ ~~[2.07]~~ 2.00 square inches. Insert: 3.4-30-01 | 8 hours |
| <3.1.A.8.c.1> <DOC A.9> | ~~OR~~ F. LTOP ~~System~~ inoperable for any reason other than Condition A, ~~[B,]~~ C, D, ~~E, or F~~ or E | Insert: 3.4-30-02 | |

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | | FREQUENCY |
|---|---|---|---|
| <DOC M.2> | SR 3.4.12.1 | Verify ~~a maximum of [one] [HPI] pump is~~ no HHSI pumps are capable of injecting into the RCS. | 12 hours |
| | ~~SR 3.4.12.2~~ | ~~Verify a maximum of one charging pump is capable of injecting into the RCS.~~ | ~~12 hours~~ |
| <DOC M.1> | SR 3.4.12.~~3~~ 2 Insert: 3.4-30-04 | Verify each accumulator ~~is isolated~~ Insert: 3.4-30-03 | 12 hours |

(continued)

**INSERT: 3.4-30-01**

| | | | |
|---|---|---|---|
| | | OR | |
| <3.1.A.8.c.2> <DOC LA.4> | E.2 | Increase average RCS cold leg temperature to ≥ 319°F. | 8 hours |
| | | OR | |
| <3.1.A.8.c.3> <DOC M.5> | E.3 | Verify pressurizer level, RCS pressure, and RCS injection capability are within limits specified in PTLR for OPS not OPERABLE. | 8 hours<br>AND<br>Once per 12 hours thereafter |

**INSERT: 3.4-30-02**

| | | | |
|---|---|---|---|
| <3.1.A.8.c> | F.1 | Depressurize RCS and establish RCS vent of ≥ 2.00 square inches. | 8 hours |

**INSERT: 3.4-30-03**

discharge isolation valve is closed and de-energized;

**INSERT: 3.4-30-04**

| | | |
|---|---|---|
| | OR | |
| <DOC M.1> | Verify each accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. | 12 hours |

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY | |
|---|---|---|---|
| ~~SR 3.4.12.4~~ | ~~Verify RHR suction valve is open for each required RHR suction relief valve.~~ | ~~12 hours~~ | |
| ⟨Doc M.6⟩ SR 3.4.12.~~5~~ 3 | ----------NOTE---------- <br>Only required to be ~~performed~~ met when complying with LCO 3.4.12.b. <br>------------------------------ <br><br>Verify RCS vent ≥ ~~[2.07]~~ 2.00 square inches ~~open~~ established. | 12 hours for unlocked open vent valve(s) <br><br>**AND** <br><br>31 days for locked open vent valve(s) | PA.1 |
| Insert: 3.4-31-01 | | | |
| ⟨Doc M.7⟩ ⟨Doc A.6⟩ SR 3.4.12.~~6~~ 5 | Verify PORV block valve is open for each required PORV. | 72 hours | |
| ~~SR 3.4.12.7~~ | ~~Verify associated RHR suction isolation valve is locked open with operator power removed for each required RHR suction relief valve.~~ | ~~31 days~~ | |
| ⟨Doc L.1⟩ SR 3.4.12.~~8~~ 6 | ----------NOTE---------- <br>Not required to be met until 12 hours after decreasing RCS average cold leg temperature to ~~≤~~ < ~~[275]~~ 319°F. <br>------------------------------ <br><br>⟨Table 4.1-1, Item 38⟩ Perform a COT on each required PORV, excluding actuation. | ~~31 days~~ 24 months | CLB.1 |

(continued)

**INSERT 3.4-31-01:**

| | | |
|---|---|---|
| SR 3.4.12.4<br><br><T4.1-1, Item 38><br><DOC A.10> | met<br>-------------------NOTE-------------------<br>Only required to be performed when complying with LCO 3.4.12.a.<br>-----------------------------------------<br><br>Perform CHANNEL CHECK of Overpressure Protection (OPS) instrument channels. | 24 hours |

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|---|
| ~~SR 3.4.12.9 Perform CHANNEL CALIBRATION for each required PORV actuation channel.~~ | ~~[18] months~~ |

Insert: 3.4-32-01

Insert: 3.4-32-02

Insert: 3.4-32-03

**INSERT 3.4-32-01:**

| | | |
|---|---|---|
| SR 3.4.12.7 | Perform CHANNEL CALIBRATION ~~of~~ for each required OPS channel as follows: | |
| ⟨Table 4.1-1 Item 38⟩ | a. OPS actuation channels; and | 18 months |
| | b. RCS pressure and temperature instruments. | 24 months |

**INSERT 3.4-32-02:**

| | | | |
|---|---|---|---|
| | SR 3.4.12.8 | -------------------NOTES------------------- | |
| <3.1.A.1.h> | | 1. Not required to be met when average RCS cold leg temperature is ≥ 319°F. | |
| <3.1.A.1.h> | | 2. Not required to be met if SR 3.4.12.9 is met.<br>------------------------------------------- | |
| <3.1.A.1.h><br><DOC M.4> | | Verify each of the following conditions are satisfied prior to starting any RCP: | Within 15 minutes prior to starting any RCP |
| <3.1.A.1.h.1><br><3.1.A.1.h.3> | | a. Secondary side water temperature of the hottest steam generator (SG) is less than or equal to the coldest RCS cold leg temperature; and | |
| <3.1.A.1.h> | | b. RCS makeup is less than or equal to RCS losses; and | |
| <3.1.A.1.h.1><br><3.1.A.1.h.3> | | c. Steam generator pressure is not decreasing; and | |
| <3.1.A.1.h.1> | | d.1 Overpressure Protection System (OPS) is OPERABLE;<br><br><u>OR</u> | |
| <3.1.A.1.h.3> | | d.2.1 RCS pressure less than nominal OPS setpoint specified in the PTLR; and | |
| <3.1.A.1.h.3><br><DOC M.3> | | d.2.2 Pressurizer level, RCS pressure, and RCS injection capability are within limits specified in PTLR for OPS not OPERABLE. | |

INSERT 3.4-32-03:

| | | |
|---|---|---|
| SR 3.4.12.9 | -------------------NOTES---------------------- | |
| <3.1.A.1.h> | 1. Not required to be met when average RCS cold leg temperature is ≥ 319°F. | |
| <3.1.A.1.h> | 2. Not required to be met if SR 3.4.12.8 is met.<br>------------------------------------------------ | |
| <3.1.A.1.h><br><DOC H.4> | Verify each of the following conditions are satisfied prior to starting any RCP: | Within 15 minutes prior to starting any RCP |
| <3.1.A.1.h.2> | a. Secondary side water temperature of the hottest steam generator is ≤ 64°F above the coldest RCS cold leg temperature; and | |
| <3.1.A.1.h> | b. RCS makeup is less than or equal to RCS losses; and | |
| <3.1.A.1.h.2> | c. Overpressure Protection System (OPS) is OPERABLE; and | |
| <3.1.A.1.h.2> | d. Pressurizer level is ≤ 73%; and | |
| <3.1.A.1.h.2> | e. Coldest RCS cold leg temperature is within limits specified in PTLR for RCP start with OPS OPERABLE and SG temperature greater than RCS cold leg temperature. | |

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.12 Low Temperature Overpressure Protection (LTOP) ~~System~~

BASES

BACKGROUND

is established to limit

~~The~~ LTOP ~~System controls~~ RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

Insert: B 3.4-58-01

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

because

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

limiting maximum

Insert: B 3.4-58-02

power operated

PORV

sufficient to provide

(PORVs)

This LCO provides RCS overpressure protection by ~~having a~~ minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all ~~but [one]~~ [high pressure injection (HPI)] pump [and one charging pump] incapable of injection into the RCS and isolating the accumulators. The pressure relief capacity requires either two redundant RCS relief valves ~~or~~ a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is~~, the~~ overpressure protection ~~device that acts~~ to terminate an increasing pressure event.

Insert: B 3.4-12-58-03

(continued)

**INSERT: B 3.4-58-01**

When the RHR System is isolated from the RCS, the RHR System is protected from overpressure by two spring loaded relief valves (SI-733A and SI-733B). When the RHR System is not isolated from the RCS, the RHR System is protected from overpressure by spring loaded relief valve (i.e., AC-1836) which has sufficient capacity to accommodate all 3 charging pumps. However, this relief valve does not have sufficient capacity to ensure that the RHR system does not exceed design pressure limits during a mass addition resulting from an inadvertent injection of one or more high head safety injection (HHSI) pumps. Therefore, LTOP requirements are used to protect the RHR System whenever the RHR System is not isolated from the RCS.

**INSERT: B 3.4-58-02**

is achieved by not permitting any High Head Safety Injection (HHSI) pumps to be

**INSERT: B 3.4-58-03**

Alternately, if redundant PORVs are not Operable or an RCS vent cannot be established, LTOP protection may be established by limiting the pressurizer level to within limits specified in the PTLR consistent with the number of charging pumps and number of high head safety injection (HHSI) pumps capable of injecting into the RCS. This approach is acceptable because pressurizer level can be maintained such that it will either accommodate any anticipated pressure surge or allow operators time to react to any unanticipated pressure surge. When pressurizer level is used to satisfy LTOP requirements, operator action is assumed to terminate the unplanned HHSI pump injection within 10 minutes.

BASES

high pressure

limited

BACKGROUND (continued)

With ~~minimum~~ coolant input capability, the ability to ~~provide core~~ coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one ~~[HPI or] charging~~ pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

Create an overpressure condition by

Insert: B 3.4-59-01

HHSI

Insert: B 3.4-59-02

can

either the OPS

~~The~~ LTOP ~~System~~ for pressure relief ~~consists~~ of ~~two PORVs with reduced lift settings, or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve,~~ or a depressurized RCS and an RCS vent of sufficient size. Two ~~RCS relief valves~~ are required for redundancy. One ~~RCS relief valve~~ has adequate relieving capability to keep from overpressurization for the required coolant input capability.

PORVs

PORV

Insert: B 3.4-59-03

PORV Requirements

Insert: B 3.4-59-04

~~As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The LTOP actuation logic monitors both RCS temperature and RCS pressure and determines when a condition not acceptable in the PTLR limits is approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.~~

~~The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.~~

~~The PTLR presents the PORV setpoints for LTOP. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.~~

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.4.12 - Low Temperature Overpressure Protection (LTOP)

**INSERT: B 3.4-59-01**

There is no restriction on the status of charging pumps when LTOP is established using either a PORV or an RCS vent.

**INSERT: B 3.4-59-02**

Charging pumps and low pressure injection systems are available to provide makeup even when LTOP requirements are applicable.

**INSERT: B 3.4-59-03**

When configured to provide low temperature overpressure protection, the PORVs are part of the Overpressure Protection System (OPS).

**INSERT: B 3.4-59-04**

The Overpressure Protection System (OPS) provides the low temperature overpressure protection by controlling the Power Operated Relief Valves (PORVs) and their associated block valves with pressure setpoints that vary with RCS cold leg temperature. Specifically, cold leg temperature signals from three RCS loops are supplied to three associated function generators that calculate the maximum RCS pressures allowed at those temperatures. The maximum RCS pressure limits at any RCS temperature correspond to the 10 CFR 50, Appendix G, limit curve maintained in the Pressure and Temperature Limits Report and are used as the OPS pressure setpoint. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

In addition to generating the OPS pressure setpoint, the same cold leg temperature signals are used to "arm" the OPS when RCS temperature falls below the temperature at which low temperature overpressure protection is required (319°F). Each PORV opens when a two-out-of-two (temperature and pressure) coincidence logic is satisfied. OPS is "armed" when RCS temperature falls below the temperature that satisfies one half of the two-out-of-two (temperature-pressure) coincidence logic. When OPS is enabled, the PORVs will open if RCS pressure exceeds the calculated pressure setpoint that varies with RCS temperature. The PORV block valves open when the RCS temperature falls below the OPS arming temperature. Note that the control switches for the PORV and PORV block valves must be in the AUTO position and the OPS states links closed for OPS signals to actuate the PORVs.

Three channels of RCS cold leg temperature are used in the two-out-of-three coincidence logic to satisfy the temperature portion of the two-out-of-two (temperature and pressure) coincidence logic for each PORV. Three channels of RCS pressure are used in a two-out-of-three coincidence logic to satisfy the pressure portion of the two-out-of-two (temperature-pressure) coincidence logic for each PORV. Use of a two-out-of-three coincidence logic for pressure and for temperature ensures that a single failure will not cause or prevent an OPS actuation. Use of two PORVs, each with adequate relieving capability to prevent overpressurization, ensures that a single failure will not prevent an OPS actuation.

BASES

BACKGROUND

PORV Requirements (continued)

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

~~RHR Suction Relief Valve Requirements~~

~~During LTOP MODES, the RHR System is operated for decay heat removal and low pressure letdown control. Therefore, the RHR suction isolation valves are open in the piping from the RCS hot legs to the inlets of the RHR pumps. While these valves are open and the RHR suction valves are open, the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.~~

~~The RHR suction isolation valves and the RHR suction valves must be open to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. Autoclosure interlocks are not permitted to cause the RHR suction isolation valves to close. The RHR suction relief valves are spring loaded, bellows type water relief valves with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.~~

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

Insert: B 3.4-60-01

~~For an RCS vent to meet the flow capacity requirement, it requires removing a pressurizer safety valve, removing a PORV's internals,~~ and disabling its block valve in the open

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.4.12 - Low Temperature Overpressure Protection (LTOP)

**INSERT: B 3.4-60-01**

Multiple methods exist for establishing the required RCS vent capacity including removing or blocking open a PORV

BASES

BACKGROUND

RCS Vent Requirements (continued)

Insert: B3.4-61-01

position~~, or similarly establishing a vent by opening an RCS vent valve~~. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

3

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, ~~and in MODE 4~~ with RCS cold leg temperature exceeding ~~[275]~~ °F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At ~~about [275]~~ °F and below, overpressure prevention falls to two OPERABLE ~~RCS relief valves~~ or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability.

411

319

Insert: B3.4-61-02

Insert: B3.4-61-03

Insert: B3.4-61-04

OPS (PORVs)

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, ~~the~~ LTOP ~~System~~ must be re-evaluated to ensure its functional requirements can still be met using the ~~RCS relief valve method~~ or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

3

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

(continued)

**INSERT: B 3.4-61-01**

An RCS vent of ≥ 2.00 square inches when no HHSI pump is capable of injecting into the RCS; or, an RCS vent with opening greater than or equal to one pressurizer code safety valve flange and up to two HHSI pumps capable of injecting into the RCS will satisfy LTOP requirements because either configuration ensures pressure limits are not exceed during a transient. Alternately, an RCS vent of ≥ 2.00 square inches coupled with a pressurizer level ≤ 0% and up to two HHSI pumps capable of injecting into the RCS will satisfy LTOP requirements because it ensures a minimum of 10 minutes for operator action before pressure limits are exceeded during a transient.

**INSERT: B 3.4-61-02**

PORVs in conjunction with the Overpressure Protection System (OPS)

**INSERT: B 3.4-61-03**

Alternately, if redundant PORVs are not Operable, Low Temperature Overpressure protection may be maintained by limiting the pressurizer level to within limits specified in the PTLR consistent with the number of charging pumps and number of high head safety injection (HHSI) pumps capable of injecting into the RCS. This approach is acceptable because pressurizer level can be established to either accommodate any anticipated pressure surge or allow operators time to react to any unanticipated pressure surge.

**INSERT: B 3.4-61-04**

When the RCS temperature is greater than the LTOP arming temperature (i.e., ≥ 319°F) but below the minimum temperature at which the pressurizer safety valves lift prior to violation of the 10 CFR 50, Appendix G, limits (i.e., ≤ 411°F), administrative controls in the Technical Requirements Manual (TRM) (Ref. 4) are used to limit the potential for exceeding 10 CFR 50, Appendix G, limits. These administrative controls may include operating with a bubble in the pressurizer and/or otherwise limiting plant time or activities when the RCS temperature is in the specified range. The use of administrative controls to govern operation above the LTOP arming temperature but below the minimum temperature at which the pressurizer safety valves lift prior to violation of the 10 CFR 50, Appendix G, limits is consistent with the guidance provided in Generic Letter 88-011, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations (Ref.2). GL 88-011 states that automatic, or passive, protection of the P-T limits will not be required but administratively controlled when in the upper end of the 10 CFR 50, Appendix G, temperature range.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

HHSI pumps

a. Rendering all ~~but [one] [HPI] pump [and one charging pump]~~ incapable of injection;

Insert: B3.4-62-01

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

Insert: B3.4-62-02

c. Disallowing start of an RCP ~~if secondary temperature is more than [50]°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops—MODE 4," and LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," provide this protection~~.

3

POPVs

Insert: B3.4-62-03

Insert: B3.4-62-04

The Reference ~~4~~ analyses demonstrate that either one ~~RCS relief valve~~ or the depressurized RCS and RCS vent can maintain RCS pressure below limits ~~when only one [HPI] pump [and one charging pump are] is [are] actuated. Thus, the LCO allows only [one] [HPI] pump [and one charging pump] OPERABLE during the LTOP MODES.~~ ~~Since neither one RCS relief valve nor the RCS vent can handle~~ therefore, the pressure transient need from accumulator injection, when RCS temperature is low, the LCO also requires the accumulators isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

If the accumulators are isolated and not depressurized, then

~~The isolated~~ accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. ~~The analyses show the effect of accumulator discharge is over a narrower RCS temperature range ([175]°F and below) than that of the LCO ([275]°F and below).~~

(continued)

**INSERT: B 3.4-62-01**

or maintaining accumulator pressure less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR;

**INSERT: B 3.4-62-02**

unless conditions are established that ensure a RCP pump start will not cause a pressure excursion that will exceed LTOP limits. Required conditions for starting a RCP when LTOP is required include a combination of primary and secondary water temperature differences and Overpressure Protection System (OPS) status or pressurizer level. Meeting the LTOP RCP starting surveillances ensures that theses conditions are satisfied prior to an RCP pump start.

**INSERT: B 3.4-62-03**

when no HHSI pump is capable of injecting into the RCS. This assumes an RCS vent of ≥ 2.00 square inches. The same protection can be provided when up to two HHSI pumps are capable of injecting into the RCS assuming an RCS vent with opening greater than or equal to one code pressurizer safety valve flange. Alternately, LTOP requirements can be satisfied by various combinations of pressurizer level, RCS pressure, and RCS injection capability (i.e., maximum number of HHSI pumps and/or charging pumps) shown in the PTLR. These combinations of pressurizer level, RCS pressure, and RCS injection capability satisfy LTOP requirements by ensuring a minimum of 10 minutes for operator action to terminate an unplanned event prior to exceeding maximum allowable RCS pressure.

**INSERT: B 3.4-62-04**

None of the analyses addressed

BASES

APPLICABLE SAFETY ANALYSES

Heat Input Type Transients (continued)

Fracture mechanics analyses established the temperature of LTOP Applicability at ~~[275]~~ 319°F.

The consequences of a ~~small break~~ loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. ~~5 and 6~~ 5 and 6), requirements by ~~having a maximum of [one] [HPI]/pump [and one charging pump] OPERABLE and SI actuation enabled~~.

Insert: B 3.4-63-01

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient ~~of [one] [HPI] pump [and one charging pump]~~ with HHSI not injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

Insert: B 3.4-63-02

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

(continued)

**INSERT: B 3.4-63-01**

having ECCS OPERABLE in accordance with requirements in LCO 3.5.3, ECCS-Shutdown.

**INSERT: B 3.4-63-02**

The OPS setpoint is based on a comparative analysis of Reference 3, with allowances for metal/fluid temperature differences, static head due to elevation differences, and dynamic head from the operation of the reactor coolant pumps and RHR pumps.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~[RHR Suction Relief Valve Performance]~~

~~The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs. Analyses must show that one RHR suction relief valve with a setpoint at or between [436.5] psig and [463.5] psig will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting LTOP event, an RHR suction relief valve will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation ≤ 10% of the rated lift setpoint.~~

~~Although each RHR suction relief valve may itself meet single failure criteria, its inclusion and location within the RHR System does not allow it to meet single failure criteria when spurious RHR suction isolation valve closure is postulated. Also, as the RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for LTOP.~~

~~The RHR suction relief valves are considered active components. Thus, the failure of one valve is assumed to represent the worst case single active failure.~~

RCS Vent Performance

1.4

With the RCS depressurized, analyses show a vent size of 2.07 square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, ~~[one] HPI pump [and one charging pump] OPERABLE,~~ maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

Insert: B3.4-64-01

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

Insert: B3.4-64-02

The RCS vent is passive and is not subject to active failure.

(continued)

**INSERT: B 3.4-64-01**

assuming no HHSI pump and no accumulator injects into the RCS. The LCO limit for an RCS vent is conservatively established at 2.00 square inches.

**INSERT: B 3.4-64-02**

An RCS vent with opening greater than or equal to one pressurizer code safety valve flange and up to two HHSI pumps capable of injecting into the RCS will satisfy LTOP requirements because it ensures pressure limits are not exceeded during a transient. An RCS vent of ≥ 2.00 square inches coupled with a pressurizer level ≤ 0% and up to two HHSI pumps capable of injecting into the RCS will satisfy LTOP requirements because it ensures a minimum of 10 minutes for operator action before pressure limits are exceeded during a transient.

BASES

APPLICABLE SAFETY ANALYSES

RCS Vent Performance (continued)

~~The~~ LTOP ~~System~~ satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires that no HHSI pumps be ~~one [HPI] pump [and one charging pump]~~ capable of injecting into the RCS and all accumulator discharge isolation valves closed and ~~immobilized~~ de-energized if ~~when~~ accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. ~~Two RCS relief valves, as follows:~~

~~1.~~ Two OPERABLE PORVs; or

Insert: B3.4-65-01

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

Insert: B3.4-65-02

~~[2. Two OPERABLE RHR suction relief valves; or]~~

~~[An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valve and its RHR suction valve are open, its setpoint is at or between [436.5] psig and [463.5] psig, and testing has proven its ability to open at this setpoint.]~~

(continued)

**INSERT: B 3.4-65-01**

configured as part of an OPERABLE Overpressure Protection System (OPS)

**INSERT: B 3.4-65-02**

The OPS is OPERABLE for LTOP when there are three OPERABLE RCS pressure channels and three OPERABLE RCS temperature channels. The OPS is still OPERABLE when an inoperable RCS pressure or temperature channel is in the tripped condition.

BASES

LCO (continued)

3. ~~One~~ OPERABLE ~~PORV~~ and ~~one~~ OPERABLE ~~RHR~~ suction ~~relief valve~~; or

b. A depressurized RCS and an RCS vent.

indent

An RCS vent is OPERABLE when open with an area of ≥ [2.07] square inches.

2.00

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

Insert: B3.4-66-01

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ [275]°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above [275]°F. When the reactor vessel head is off, overpressurization cannot occur.

average

<

Insert: B3.4-66-02

319

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above [275]°F.

Insert: B3.4-66-03

319

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

The Applicability is modified by ~~a Note stating~~ that accumulator isolation is only required when the accumulator pressure is more than ~~or at~~ the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

three Notes. Note 1 states

Insert: B3.4-66-04

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.4.12 - Low Temperature Overpressure Protection (LTOP)

**INSERT: B 3.4-66-01**

This LCO is applicable whenever the RHR System is not isolated from the RCS to protect the RHR system piping. When average RCS cold leg temperatures are ≥ 319°F, RHR system piping is adequately protected by making the accumulators and all HHSI pumps incapable of injecting into the RCS. Therefore, a Note in the LCO specifies that requirements for the OPS System and/or an RCS vent are not Applicable when average RCS cold leg temperature is ≥ 319°F.

**INSERT: B 3.4-66-02**

to provide protection for the RCS pressure boundary

**INSERT: B 3.4-66-03**

Although LTOP is not Applicable when the RCS temperature is greater than the LTOP arming temperature (i.e., ≥ 319°F) but below the minimum temperature at which the pressurizer safety valves lift prior to violation of the 10 CFR 50, Appendix G, limits (i.e., ≤ 411°F), administrative controls in the Technical Requirements Manual (TRM) (Ref. 4) are used to limit the potential for exceeding 10 CFR 50, Appendix G, limits.

**INSERT: B 3.4-66-04**

Note 2 ensures that LCO 3.4.12 will not prohibit a HHSI pump being energized and aligned to the RCS as needed to support emergency boration or to respond to a loss of RHR cooling.

Note 3 specifies that one HHSI pump may be made capable of injecting into the RCS for a period not to exceed 8 hours to perform pump testing. During testing, administrative controls are used to ensure that HHSI testing will not result in exceeding RCS or RHR system pressure limits.

BASES (continued)

ACTIONS

Insert: B 3.4-67-01

~~A.1 [and B.1]~~

~~With two or more HPI pumps capable of injecting into the RCS, RCS overpressurization is possible.~~

~~To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.~~

~~Required Action B.1 is modified by a Note that permits two charging pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps.~~

B C C
~~C~~.1, ~~D~~.1, and ~~D~~.2

Insert: B 3.4-67-02

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, ~~Required Action D.1 and Required Action D.2~~ C provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to ~~> [275]~~ ≥ 319°F, ~~an accumulator pressure of [600]~~ 700 psig cannot exceed the LTOP limits if the accumulators are ~~fully~~ injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

Insert: B 3.4-67-03

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

D ~~E~~.1

~~In MODE 4~~ When ~~any~~ average RCS cold leg temperature is ~~≤ [275]~~ < 319°F, with one required ~~RCS relief valve~~ PORV inoperable, the ~~RCS relief valve~~ PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two ~~RCS relief valves [in any combination of the PORVS and the RHR suction relief valves]~~ PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

(continued)

**INSERT: B 3.4-67-01**

A.1, A.2.1, A.2.2, A.2.3, A.3.1 and A.3.2

When one or more HHSI pumps are capable of injecting into the RCS, LTOP assumptions regarding limits on mass input capability may not be met. Therefore, immediate action is required to limit injection capability consistent with the LTOP analysis assumptions and the existing combination of pressurizer level and RCS venting capacity. Required Action A.1 requires restoration with LCO requirements. Required Actions A.2 and A.3 require verification and periodic re-verification that alternate LTOP configurations are met. The Completion Times of immediately reflects the urgency that one of the acceptable LTOP configurations is established as soon as possible.

**INSERT: B 3.4-67-02**

To be considered isolated, an accumulator must have its discharge valves closed and the valve power supply breakers fixed in the open position.

**INSERT: B 3.4-67-03**

Additionally, the RHR System must be isolated from the RCS to protect RHR piping from a potential mass addition event.

BASES

ACTIONS

D ~~E~~.1 (continued)

The Completion Time considers the facts that only one of the ~~RCS relief valves~~ PORVs is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

~~F.1~~

~~The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS relief valves inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.~~

~~The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve to protect against overpressure events.~~

E ~~G~~.1

Insert B3.4-68.01

~~The RCS must be depressurized and a vent must be established within 8 hours when:~~

~~a. Both required RCS relief valves are inoperable; or~~

~~b. A Required Action and associated Completion Time of Condition A, [B,] D, E, or F is not met; or~~

~~c. The LTOP System is inoperable for any reason other than Condition A, [B,] C, D, E, or F.~~

Insert: B3.4-68.02

~~T~~he vent must be sized ≥ ~~[2.07]~~ 2.00 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.4.12 - Low Temperature Overpressure Protection (LTOP)

**INSERT: B 3.4-68-01**

When both required PORVs are inoperable or the Required Action and associated Completion Time of Condition C or D is not met, an alternate method of low temperature overpressure protection must be established within 8 hours. The acceptable alternate methods of LTOP include the following:

a. Depressurize the RCS and establish an RCS vent path; or

b. Increase average RCS cold leg temperatures to ≥ 319°F; or

c. Establish a combination of pressurizer level, RCS pressure, and RCS injection capability within limits specified in PTLR for OPS not OPERABLE. This combination will ensure at least 10 minutes for operator intervention to prevent overpressurization following a transient.

**INSERT: B 3.4-68-02**

If the option selected is to depressurize the RCS and establish an RCS vent path,

BASES

ACTIONS

G.1 ~~(continued)~~

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

Insert: B3.4-69-01

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, ~~SR 3.4.12.2~~ and SR 3.4.12.~~3~~ 2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, ~~a maximum of [one] [HPI] pump [and a maximum of one charging pump]~~ all HHSI pumps are verified incapable of injecting into the RCS. ~~and the accumulator discharge isolation valves are verified closed and locked out.~~

Insert: B3.4-69-02

The ~~[HPI]~~ HHSI pump~~[s]~~ ~~and charging pump[s]~~ are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. ~~An alternate method of LTOP control~~ Other methods may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in ~~[pull to lock]~~ Trip Pullout and at least one valve in the discharge flow path being closed.

Insert: B3.4-69-03

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

~~SR 3.4.12.4~~

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.7 for the RHR suction isolation valve Surveillance.) This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.~~

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.4.12 - Low Temperature Overpressure Protection (LTOP)

**INSERT: B 3.4-69-01**

F.1

If LTOP requirements are not met for reasons other than Conditions A, B, C, D or E, LTOP requirements must be re-established by depressurizing the RCS and establishing RCS vent of ≥ 2.00 square inches within 8 hours.

**INSERT: B 3.4-69-02**

Additionally, the

**INSERT: B 3.4-69-03**

or the accumulator pressure less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.4 (continued)

The RHR suction valve is verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valve remains open.

The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.~~5~~ 3

The RCS vent of ≥ ~~2.07~~ 2.00 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that ~~cannot be~~ is not locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve, PORV, or manway cover fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO ~~3.4.12b~~ 3.4.12.b.

Insert: B 3.4-70-01

SR 3.4.12.~~6~~ 5

The PORV block valve Insert: B 3.4-70-02 must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve ~~must~~ may be remotely verified open in the ~~main~~ control room. ~~This Surveillance is performed~~ only if the PORV ~~satisfies the LCO.~~ Insert: B 3.4-70-03

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation. Insert: B 3.4-70-04

(continued)

**INSERT: B 3.4-70-01**

SR 3.4.12.4

Performance of the CHANNEL CHECK of the Overpressure Protection System (OPS) RCS pressure and temperature channels every 24 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels. This SR is required only when LCO 3.4.12.a is used to establish LTOP protection.

**INSERT: B 3.4-70-02**

opens automatically when RCS cold leg temperature is below the OPS arming temperature; however, the valves

**INSERT: B 3.4-70-03**

is being used to satisfy LCO 3.4.12.a.

**INSERT: B 3.4-70-04**

If closed, the block valve must be de-energized to prevent the valve from re-opening automatically.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.~~6~~ 5 (continued)

Insert: B3.4-71-01

The 72 hour Frequency is considered adequate ~~in view of~~ other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

~~SR 3.4.12.7~~

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valve are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.4 for the RHR suction valve Surveillance and for a description of the requirements of the Inservice Testing Program.) This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.~~

~~Every 31 days the RHR suction isolation valve is verified locked open, with power to the valve operator removed, to ensure that accidental closure will not occur. The "locked open" valve must be locally verified in its open position with the manual actuator locked in its inactive position. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve position.~~

SR 3.4.12.~~8~~ 6

Performance of a COT is required within 12 hours after decreasing RCS temperature to ~~≤ [275]~~ < 319 °F and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

The ~~12 hour~~ 31 day Frequency considers ~~the unlikelihood of a low temperature overpressure event during this time.~~ Insert: B3.4-71-02

A Note has been added indicating that this SR is required to be met 12 hours after decreasing RCS cold leg temperature to ~~≤ [275]~~ < 319 °F. The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP

(continued)

**INSERT: B 3.4-71-01**

because the PORV block valves are opened automatically by the OPS when below the OPS arming temperature if the valve control is positioned to auto and

**INSERT: B 3.4-71-02**

the demonstrated reliability of the Overpressure Protection System and the PORVs.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.~~8~~ 6 (continued)

setting. The test must be performed within 12 hours after entering the LTOP MODES.

SR 3.4.12.~~9~~ 7

Insert: B 3.4-72-01

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every [18] months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ~~Generic Letter 88-11~~ Insert: B 3.4-72-03
3. ~~ASME, Boiler and Pressure Vessel Code, Section III.~~
4. ~~FSAR, Chapter [15]~~ Insert: B 3.4-72-04
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. ~~Generic Letter 90-06.~~
8. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

Insert: B 3.4-72-02

**INSERT: B 3.4-72-01**

Performance of a CHANNEL CALIBRATION of RCS pressure and temperature instruments that support the Overpressure Protection System is required every 24 months. These calibrations verify both the OPS and PORV function and ensure the OPERABILITY of

**INSERT: B 3.4-72-02**

SR 3.4.12.8 and SR 3.4.12.9

The RCP starting prerequisites must be satisfied prior to starting or jogging any reactor coolant pump (RCP) when low temperature overpressure protection is required. The RCP starting prerequisites prevent an overpressure event due to thermal transients when an RCP is started. Plant conditions prior to the RCP start determines whether SR 3.4.12.8 or SR 3.4.12.9 must be satisfied prior to starting any RCP.

The principal contributor to an RCP start induced thermal and pressure transient is the difference between RCS cold leg temperatures and secondary side water temperature of any SG prior to the start of an RCP. The RCP starting prerequisites vary depending on plant conditions but include the following: reactor coolant temperature relative to the LTOP arming temperature; secondary side water temperature of the hottest SG relative to the temperature of the coldest RCS cold leg temperature; and, the status of the Overpressure Protection System (OPS). When the OPS is inoperable, additional compensatory requirements are required including limits for the pressurizer level and RCS pressure and temperature. When a pressurizer level is specified as a requirement, the level specified is sufficient to prevent the RCS from going water solid for 10 minutes which is sufficient time for operator action to terminate the pressure transient.

SR 3.4.12.8 is used if secondary side water temperature of the hottest steam generator (SG) is less than or equal to the coldest RCS cold leg temperature. SR 3.4.12.9 is more restrictive and is used if the secondary side water temperature of the hottest steam generator is ≤ 64°F above the coldest RCS cold leg temperature. RCP starting is prohibited if the hottest steam generator is > 64°F above RCS cold leg temperature or if neither of the RCP starting prerequisites SRs can be satisfied.

**INSERT: B 3.4-72-02** (continued)

The steam generator temperature may be measured using the Control Room instrumentation or, as a backup, from a contact reading off the steam generator's shells. Pressurizer level may be determined using control room instrumentation or alternate methods.

The FREQUENCY of the RCP starting prerequisites SRs is Within 15 minutes prior to starting any RCP. This means that each of the required verifications must be performed within 15 minutes prior to the pump start and must be met at the time of the pump start.

SR 3.4.12.8 and SR 3.4.12.9 are each modified by two Notes. Note 1 specifies that these SRs are required as a condition for pump starting only when the RCS is below the LTOP arming temperature. Note 2 specifies that meeting either SR 3.4.12.8 or SR 3.4.12.9 ensures that pump starting prerequisites are met.

**INSERT: B 3.4-72-03**

2. Generic Letter 88-011, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations.

**INSERT: B 3.4-72-04**

3. IP3 Low Temperature Overpressurization System Analysis Final Report, August 24, 1984, in conjunction with ASME Code Case N-514, Low Temperature Overpressure Protection, February 12, 1992.

4. IP3 Technical Requirements Manual.

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Conversion Package

# Technical Specification 3.4.12:
# "Low Temperature Overpressure Protection (LTOP)"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

## JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
## ITS SECTION 3.4.12 - Low Temperature Overpressure Protection (LTOP)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 NUREG-1431, Rev 1, Section 3.4.12, was modified as needed to reflect the IP3 design and current licensing basis. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

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# Technical Specification 3.4.13:
# "RCS Operational LEAKAGE"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and

e. 432 gallons per day primary to secondary LEAKAGE through any one SG.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE. | A.1 Reduce LEAKAGE to within limits. | 4 hours |
| B. Required Action and associated Completion Time of Condition A not met.<br><br>OR<br><br>Pressure boundary LEAKAGE exists. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 5. | 6 hours<br><br><br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.13.1 | ------------------NOTE------------------<br>Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation.<br>----------------------------------------<br><br>Verify RCS Operational leakage is within limits by performance of RCS water inventory balance. | -----NOTE------<br>Only required to be performed during steady state operation<br>--------------<br><br>72 hours |
| SR 3.4.13.2 | Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program. | In accordance with the Steam Generator Tube Surveillance Program |

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.13 RCS Operational LEAKAGE

### BASES

**BACKGROUND**

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE. 10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

## APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for events resulting in steam discharge to the atmosphere assumes a range of primary to secondary LEAKAGE from 0.1 gpm to 10 gpm as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 2) analysis for SGTR assumes the contaminated secondary fluid is released via safety valves and atmospheric dump valves. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes a range of primary to secondary LEAKAGE as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 and the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36.

## LCO

RCS operational LEAKAGE shall be limited to:

a. <u>Pressure Boundary LEAKAGE</u>

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of

BASES

LCO

a. (continued)

this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount and is consistent with the capability of the equipment required by LCO 3.4.15, RCS Leakage Detection Instrumentation. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE, the leakage into closed systems or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm (1440 gpd) through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

BASES

LCO (continued)

e. Primary to Secondary LEAKAGE through Any One SG

The 432 gallons per day (0.3 gpm) limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

Leakage past PIVs or other leakage into closed systems is that leakage that can be accounted for and contained by a system not directly connected to the atmosphere. Leakage past PIVs or other leakage into closed systems is not included in the limits for either identified or unidentified LEAKAGE but PIV leakage must be within the limits specified for PIVs in LCO 3.4.14, "RCS Pressure Isolation Valves (PIV)." Leakage past PIVs or other leakage into closed systems is quantified before being exempted from the limits for identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

BASES

ACTIONS (continued)

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and blowdown systems.

The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. Therefore, this SR is not required to be performed in MODES 3

BASES

## SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

and 4 until 12 hours of steady state operation near operating pressure have been established.

Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation." It should be noted that LEAKAGE past seals and gaskets, measured leakage past PIVs, and other leakage into closed systems is not pressure boundary LEAKAGE.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. A Note under the Frequency column states that this SR is required to be performed during steady state operation.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. FSAR, Section 14.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.13:
## "RCS Operational LEAKAGE"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.1-31 | 121 | 121 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-32 | 121 | 121 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-33 | 121 | 121 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-34 | 121 | 121 | No TSCRs | No TSCRs for this Page | N/A |
| T 4.1-3(1) | 178 TSCR 97-156, 98-043 | 178 TSCR 97-156, 98-043 | IPN 98-043 | Instrument Channel Surveillance Intervals Extended to 24 Months | Incorporated |
| T 4.1-3(1) | 178 TSCR 97-156, 98-043 | 178 TSCR 97-156, 98-043 | IPN 97-156 | SR Freq for Main Turbine Stop and Control Valves | Incorporated |

LCO 3.4.13, Applicability - Mode 1, 2, 3, 4 — A.6

LCO 3.4.13

~~F.~~ **LEAKAGE OF REACTOR COOLANT**

~~Specification~~

~~1.~~ ~~If leakage of reactor coolant is indicated by the means available such as water inventory balance, monitoring equipment or direct observation a follow-up evaluation of the safety implications shall be initiated as practicable but no later than within 4 hours. Any indicated leak shall be considered to be a real leak until it is determined that the indicated leak cannot be substantiated by direct observation or other indication.~~ — A.8

LCO 3.4.13.b
Req Act A.1
Req. Act B.1, B.2

~~2.~~ If the leakage rate, ~~excluding controlled leakage sources such as the Reactor Coolant Pump Controlled Leakage Seals and Leakage into Closed Systems,~~ — A.3 exceeds 1 gpm and the source of leakage is not identified, reduce the leakage rate to within limits within four hours or be in hot shutdown within the next six hours and in cold shutdown within the ~~following~~ ~~30~~ 36 — A.1 hours. identified — L.2

LCO 3.4.13.c

~~3.~~ If the sources of leakage are identified ~~and the results of the evaluation are that continued operation is safe~~ — A.8, operation of the reactor with a ~~total~~ identified leakage, ~~other than from controlled sources or into closed systems~~ — A.3 not exceeding 10 gpm shall be permitted ~~except as specified in 3.1.F.4 below~~.

LCO 3.4.13.a
Req. Act B.1, B.2

~~4.~~ If it is determined that leakage exists through a non-isolable fault which has developed in a Reactor Coolant System Component Body, pipe wall (excluding steam generator tubes), vessel wall or pipe weld, the reactor shall be brought to ~~the cold shutdown condition within twenty-four hours~~ — L.1. Mode 3 in 6 hrs & Mode 5 in 36 hrs — L.2

LCO 3.4.13.c
Req. Act A.1, B1, B.2

~~5.~~ If the ~~total~~ identified leakage, ~~other than from controlled sources or into closed systems~~ — A.3 exceeds 10 gpm, ~~the reactor shall be placed in the hot shutdown condition within four hours and the cold shutdown condition within an additional twenty-four hours.~~ Req. Act A.1, B.1, B.2 — L3

6. ~~The reactor shall not be restarted following shutdown as per items 3.1.F.2, 3, 4, or 5, above, until the leak is repaired or until the problem is otherwise corrected.~~ — A.4

SEE ITS 5.5.8

7. Whenever the reactor is shutdown, or a steam generator removed from service, in order to investigate steam generator tube leakage and/or to plug or otherwise repair a leaking tube, the Authority shall inform the NRC before the reactor is brought critical.

LCO 3.4.13.e
LCO 3.4.13.d
Req. Act A.1, B.1, B.2

~~8.~~ Primary to secondary leakage through the steam generator tubes shall be limited to 0.3 gpm (432 gpd) per steam generator and the total leakage through all four steam generators shall be limited to 1.0 gpm (1440 gpd). With any steam generator tube leakage greater than ~~this limit the reactor shall be placed in the hot shutdown condition within four hours and the cold shutdown condition within an additional twenty-four hours.~~

Req. Act A.1, B.1, B.2 — L.4

Amendment No. ~~77~~, ~~74~~, 121

9. ~~If leakage from two or more tubes in the steam generators in any 20-day period is observed or determined, the reactor shall be brought to the hot shutdown condition within four hours and the cold shutdown condition within an additional twenty-four hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation. If two steam generator tube leaks attributable to the tube denting phenomena are observed after the reactor is in cold shutdown Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.~~ L.5

SEE ITS 3.4.15

10. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different principles capable of detecting leakage into containment shall be in operation, with one of the two systems sensitive to radioactivity. The system sensitive to radioactivity may be out-of-service for 48 hours, provided two other systems are available.

A.1

~~**Basis:**~~

~~Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.~~

~~Although some leak rates on the order of GPM may be tolerable from a dose point of view, it must be recognized that small leaks through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.~~

~~The distinction between identified and unidentified leakage in the specification is made because once the leakage source is identified, the seriousness can be easily evaluated. The strict limit of 1 gallon per minute for unidentified leakage is adopted because in the worst case the leakage source may increase with time or the coolant may impinge on or accumulate in a critical component.~~

A.1

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Watch Force. Under these conditions, an allowable primary system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also well within the capacity of one charging pump and makeup would be available even under the loss of off-site power condition.

Controlled sources of reactor coolant system leakage are sources which are designed to leak at a controlled rate. For example, the reactor coolant pump seals are controlled leakage sources. Leakage through a valve packing or a closed valve is not considered as controlled leakage. Leakage into closed systems is that leakage which can be accounted for and contained by a system not directly connected to the atmosphere. Leakage past the pressurizer safety valve seats and steam generator tube leakage are examples of reactor coolant system leakage into closed systems.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor (R-11).

b. The containment radiogas monitor (R-12).

c. The containment humidity detectors.

d. A leakage detection system which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units.

The most sensitive and rapid method for detecting small amounts of Reactor Coolant System leakage is the monitoring of the containment airborne radioactivity. Containment gaseous and particulate activity is continuously, automatically monitored. The leakage rate can be determined by the relationship of the airborne activity to the reactor coolant activity.

Measurement of the leakage rate to the containment atmosphere is also possible through humidity detection and condensation collection and measurement. However, it is expected that the containment activity method will give the initial indication of coolant leakage. The other methods will be employed primarily to confirm that leakage exists, to indicate the location of the leakage sources, and to measure the leakage rate.

As described above, the four reactor coolant leak detection systems are based on three different principles, i.e., activity, humidity and condensate flow measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the containment.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory control.

A.1

Four hours is allowed from the time of leakage detection to identify the leakage source and to measure the leakage rate. This time period is required since identification and quantification of leakage sources of less than ten gallons per minute require a careful gathering and evaluation of data and/or a visual inspection of the reactor coolant system.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by the limitation of steam generator leakage between the primary coolant system and the secondary coolant system. Cracks having a primary-to-secondary leakage less than 500 gallons per day during operation will have an adequate margin of safety against failure due to loads imposed by design basis accidents. The 500 gallon per day per steam generator limit is also consistent with the assumptions used to develop the Technical Specification limit on secondary coolant activity. Operating plants have demonstrated that primary-to-secondary leakage as low as 0.1 gpm will be detected. Leakage in excess of 432 gallons per steam generator or 1 gpm total for all four steam generators will require plant shutdown and an unscheduled eddy current inspection, during which the leaking tubes will be located and plugged.

**References**

FSAR Sections 11.2.3 and 14.2.4

Add SR 3.4.13.2



A.5

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS

SEE CTS MASTER MARKUP

| | Check | Frequency |
|---|---|---|
| 1. Control Rods | Rod drop times of all control rods | 24M |
| 2. Control Rods | Movement of at least 10 steps in any one direction of all control rods | Every 31 days during reactor critical operations |
| 3. Pressurizer Safety Valves | Set Point | 24M* |
| 4. Main Steam Safety Valves | Set Point | 24M |
| 5. Containment Isolation System | Automatic actuation | 24M |
| 6. Refueling System Interlocks | Functioning | Each refueling, prior to movement of core components |
| 7. Primary System Leakage | Evaluate (Add SR 3.4.13.1 Note) | ~~5 days/week~~ 72 hours |
| 8. Diesel Generators Nos. 31, 32 & 33 Fuel Supply | Fuel Inventory | Weekly |
| 9. Turbine Steam Stop And Control Valves | Closure | Not to exceed 6 months** |
| 10. L.P. Steam Dump System (6 lines) | Closure | Monthly |
| 11. Service Water System | Each pump starts and operates for 15 minutes (unless already operating) | Monthly |
| 12. City Water Connections to Charging Pumps and Boric Acid Piping | Temporary connections available and valves operable | 24M |

SR 3.4.13.1

A.5 L.7 L.6

SEE CTS MASTER MARKUP

* ~~Pressurizer Safety Valve setpoint test due no later than May 1996 may be deferred until the next refueling outage but no later than May 31, 1997.~~ TSCR 97-156

** The turbine steam stop and control valves shall be tested at a frequency determined by the methodology presented in WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency," as updated by Westinghouse Report, WOG-TVTF-93-17, "Update of BB-95/96 Turbine Valve Failure Rates and Effect on Destructive Overspeed Probabilities." The maximum test interval for these valves shall not exceed six months. Surveillance interval extension as per Technical Specification 1.12 is not applicable to the maximum test interval.

Amendment No. 10, 14, 43, 65, 97, 99, 125, 126, 127, 129, 133, 144, 165,

TSCR 98-043

TSCR 97-156

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.4.13: "RCS Operational LEAKAGE"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.1.F.2 and CTS 3.1.F.3 establish limits for unidentified and total (unidentified and identified) RCS Leakage and specifies that these limits do not apply to controlled leakage sources such as the reactor coolant pump controlled leakage seals and leakage into closed systems.

The definition of Leakage in ITS 1.0 defines leakage so that ITS LCO 3.4.13 limits are not applicable to controlled leakage sources such as the reactor coolant pump controlled leakage seals and leakage into closed systems. The ITS 3.4.13 Bases also include a clarification of these exceptions to RCS leakage limits. Using the ITS definition of leakage and the ITS 3.4.14 Bases to maintain an existing allowance is an administrative change with no significant adverse impact on safety.

A.4 CTS 3.1.F.6 specifies that the reactor shall not be restarted following a shutdown required for exceeding specified limits for RCS leakage until the leak is repaired or until the problem is otherwise corrected. ITS LCO 3.4.13 does not include an explicit statement of this requirement. This change is acceptable because ITS LCO 3.0.4 prohibits entry into a Mode or other specified condition in the Applicability when an LCO is not met except when specifically permitted by the ITS LCO. ITS 3.4.13 does not include any exceptions to ITS LCO 3.0.4. Therefore, deletion of CTS 3.1.F.6 is an administrative change with no adverse impact on safety.

A.5 CTS 4.9 requires a Steam Generator Tube Inspection Program. ITS 5.5.8, Steam Generator Tube Inspection Program, maintains these requirements (See ITS 5.5.8); however, ITS SR 3.4.13.2 is added to require that steam generator tube integrity shall be verified to be in accordance with the Steam Generator Tube Surveillance Program. This change is needed to tie the Steam Generator Tube Inspection Program to the appropriate Limiting Condition for Operation so that appropriate Required Actions are initiated when Steam Generator Tube Inspection Program requirements or acceptance criteria are not met. This is an administrative change with no adverse impact on safety because there is no change to the existing requirements.

A.6 CTS 3.1.F does not include an explicit statement of when the limits for RCS leakage are applicable; however, CTS 3.1.F.2, CTS 3.1.F.4 and CTS 3.1.F.5 establish an implied Applicability by requiring the plant placed in cold shutdown (Mode 5) if the limits are not met. ITS LCO 3.4.13 establishes the Applicability for RCS leakage rate limits as Modes 1, 2, 3, and 4. This is acceptable because the potential for reactor coolant pressure boundary leakage is greatest when the RCS is pressurized. In

Modes 5 and 6 leakage limits are not required because the reactor coolant pressure is lower, resulting in lower stresses and reduced potential for leakage. This is an administrative change with no adverse impact on safety because the ITS Applicability is consistent with the implied Applicability in CTS 3.1.F.2, CTS 3.1.F.4 and CTS 3.1.f.5.

A.7 CTS Table 4.1-3, Item 7, requires periodic evaluation of primary system leakage (See ITS 3.4.13, DOCs L.6 and L.7). ITS SR 3.4.13.1 maintains this requirement except that the SR 3.4.13.1 specifies this evaluation is performed using an RCS water inventory balance. This is an administrative change with no impact on safety because ITS SR 3.4.13.1 provides an explicit statement of a reasonable interpretation of the existing requirement.

A.8 CTS 3.1.F.1 specifies that if leakage of reactor coolant is indicated by water inventory balance, monitoring equipment or direct observation, then a follow-up evaluation of the safety implications shall be initiated as practicable but no later than within 4 hours. Any indicated leak shall be considered to be a real leak until it is determined that the indicated leak cannot be substantiated by direct observation or other indication. Additionally, CTS 3.1.F.3 specifies that operation may continue only if this evaluation determines continued operation is safe and leakage limits are not exceeded.

ITS LCO 3.4.13 does not include these requirements. The statement that indicated leaks shall be considered to be a real leaks until determined otherwise is not needed because any indicated leak will be compared to the acceptance criteria and action taken accordingly. Additionally, the requirement for special evaluations is eliminated because CTS 3.1.F.1 does not establish any criteria that would trigger an evaluation or the criteria for determining safety implications. This change is acceptable because ITS 3.4.13 establishes the action levels that require prompt investigation and resolution of RCS leakage because this level is potentially indicative of significant RCS boundary deterioration. These action levels are based on conservative engineering judgement. On a case basis, IP3 plant operators may initiate an evaluation of leakage rates or trends before exceeding the ITS 3.4.13 limits. Elimination of the requirement to evaluate RCS leakage before reaching the ITS 3.4.13

leakage limits is an administrative change with no impact on safety because CTS 3.1.F.1 does not establish the criteria that would trigger an evaluation or the criteria for determining safety implications.

MORE RESTRICTIVE

None

LESS RESTRICTIVE

L.1 CTS 3.1.F.4 specifies that there shall be no non-isolable fault in an RCS component body, pipe wall (excluding steam generator tubes), vessel wall or pipe weld (i.e., no pressure boundary leakage); otherwise, the reactor must be in cold shutdown (Mode 5) within 24 hours.

ITS LCO 3.4.13.a maintains the requirement for no pressure boundary leakage; however, if this requirement is not met, Condition B and associated Required Actions require the plant in Mode 3 in 6 hours and Mode 5 in 36 hours. Extending the time permitted to perform a plant cooldown when RCS pressure boundary leakage is identified is needed because the allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. Therefore, this change will reduce additional stresses on the RCS pressure boundary. This change is acceptable because extending the required completion time for plant cooldown does not significantly increase the probability that significant additional RCS deterioration will occur prior to the plant being in Mode 5. Therefore, this change has no significant adverse impact on safety.

L.2 CTS 3.1.F.2 establishes the limit for unidentified RCS Leakage at 1 gpm; and, CTS 3.1.F.3 and CTS 3.1.F.5 establish the limit for total RCS Leakage (identified plus unidentified) at 10 gpm. This combination of requirements limits RCS identified leakage to between 9 gpm and 10 gpm depending on the amount of RCS unidentified leakage.

ITS LCO 3.4.13.b establishes the limit for unidentified RCS Leakage at 1 gpm; and, ITS LCO 3.4.13.c establishes the limit for identified RCS Leakage at 10 gpm. ITS LCO 3.4.13 does not include a limit for total RCS leakage. This change is acceptable because the difference between a 10 gpm and an 11 gpm limit for total leakage (identified plus unidentified) is not significant considering that the limits are action levels that are indicative of potentially significant RCS boundary deterioration. These action levels are based on conservative engineering judgement and do not have intrinsic safety significance. Therefore, this change will not affect the timely identification and response to RCS leakage that is indicative of significant RCS pressure boundary deterioration. ~~Therefore,~~ this change has no significant adverse impact on safety.

L.3 CTS 3.1.F.5 requires that if RCS identified leakage (See ITS 3.4.13, DOC L.2) exceeds 10 gpm, then the reactor shall be placed in the hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within 28 hours. Under the same conditions, ITS LCO 3.4.13, Required Action A.1, allows 4 hours to reduce RCS leakage to within specified limits; otherwise, ITS LCO 3.4.13, Required Actions B.1 and B.2, specify that the reactor be in Mode 3 within 6 hours and Mode 5 within 36 hours. Allowing 4 hours to reduce identified leakage to within limits is needed to avoid plant shutdowns for some leakage conditions that are not indicative of impending RCS pressure boundary failure and which can be corrected while the plant is operating. Extending the time permitted to perform a plant cooldown when limits for identified leakage are exceeded is needed because the allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. These changes are acceptable because extending the required completion time for plant cooldown does not significantly increase the probability of significant additional RCS deterioration will occur prior to the plant being in Mode 5. Therefore, these changes have no significant adverse impact on safety.

L.4 CTS 3.1.F.8 limits primary to secondary leakage through the SG tubes to 0.3 gpm (432 gpd) per steam generator and the total leakage through all four steam generators to 1.0 gpm (1440 gpd). If these limits are not met, CTS 3.1.F.8 requires the plant in the hot shutdown (Mode 3) in 4 hours and cold shutdown (Mode 5) within 28 hours.

ITS LCO 3.4.13.d and ITS LCO 3.4.13.e maintain the same limits for primary to secondary leakage through the SG tubes; however, if these requirements are not met, ITS LCO 3.4.13, Required Action A.1, allows 4 hours to reduce leakage to within specified limits; otherwise, ITS LCO 3.4.13, Required Actions B.1 and B.2, specify that the reactor be in Mode 3 within 6 hours and Mode 5 within 36 hours. Extending the time permitted to perform a plant cooldown when primary to secondary leakage limits are exceeded is needed because the allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. These changes are acceptable because extending the required completion time for plant cooldown does not significantly increase the probability of significant additional SG tube deterioration prior to the plant being in Mode 5. Therefore, these changes have no significant adverse impact on safety.

L.5 CTS 3.1.F.9 requires that if leakage from two or more tubes in the steam generators in any 20 day period is observed or determined, then the reactor must be brought to the hot shutdown (Mode 3) within 4 hours and the cold shutdown (Mode 5) within 28 hours. Additionally, Nuclear Regulatory Commission (NRC) approval must be obtained before resuming reactor operations. Also, if two steam generator tube leaks attributable to the tube denting phenomena are observed after the reactor is in cold shutdown NRC approval must be obtained before resuming operations. These requirements were added to the CTS because of concerns regarding the condition of the Indian Point 3 steam generators in 1979.

ITS LCO 3.4.13 does not include these special requirements related to steam generator tube leaking. This is acceptable because the IP3 steam generators were replaced in 1989. Adequate controls to ensure the integrity of the steam generators are included in ITS 3.4.13, Operational Leakage, and ITS 5.5.8, Steam Generator Tube Surveillance Program. Therefore, this change has no significant adverse impact on safety.

L.6 CTS Table 4.1-3, Item 7, requires that primary system leakage be evaluated 5 days/week. ITS SR 3.4.13.1 maintains the requirement to evaluate RCS leakage (See ITS LCO 3.4.13, DOC A.7) except that the Frequency is reduced to once every 72 hours and the SR is required only during steady state operation. (Note that ITS LCO 3.4.15, Required Action A.1 will require an RCS water inventory balance every 24 hours whenever the RCS leakage detection instrumentation is not Operable (See ITS 3.4.15, DOC M.3)).

This change is needed and is acceptable because a 72 hour frequency is a reasonable interval to trend leakage and provide early indication of gradual RCS deterioration. Additionally, the 5 days/week Frequency can be reduced because this verification is not used for the prompt identification of rapid changes in RCS leakage rates and other methods that provide prompt and sensitive indication of significant increases in RCS leakage are available to the operators, including RCS Leakage detection instrumentation required by ITS LCO 3.4.15. Therefore, reducing the Frequency for evaluating RCS leakage using an inventory balance will not affect the timely identification and response to RCS leakage that is indicative of significant RCS pressure boundary deterioration. Therefore, this change has no significant adverse impact on safety.

L.7 CTS Table 4.1-3, Item 7, requires periodic evaluation of RCS leakage using an RCS water inventory balance in Modes 1, 2, 3 and 4 (See ITS 3.4.13, DOC A.6). ITS SR 3.4.13.1 maintains this requirement (See ITS LCO 3.4.13, DOC A.7 and L.7) except that the SR is not required to be performed in Mode 3 and 4 until there is 12 hours of steady state operation. This change is needed because it recognizes that an RCS water inventory balance requires steady state operating conditions and near operating pressure to yield valid results. Therefore, this change allows entry into Modes 3 and 4 without performing an RCS water inventory balance if not in steady state operation. This change is acceptable because the RCS water inventory balance must be completed before entering Mode 2 which ensures that an assessment of leakage is done before establishing conditions where RCS pressure boundary integrity is most important. This change will not affect the timely identification and response to RCS leakage that is indicative of significant RCS pressure boundary deterioration. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

None

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.4.13:
# "RCS Operational LEAKAGE"

## PART 4:

## No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.4.13 - RCS Operational LEAKAGE

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change extends the time permitted to perform a plant shutdown and cooldown when RCS pressure boundary leakage is identified from 24 hours to 36 hours. Extending the time permitted to perform a plant cooldown when RCS pressure boundary leakage is identified is needed because the allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. This change will not result in a significant increase in the probability of an accident previously evaluated because this change will reduce additional stresses on the RCS pressure boundary. Additionally, extending the required completion time for plant cooldown does not significantly increase the probability that significant additional RCS deterioration will occur prior to the plant being in Mode 5. This change will not result in a significant increase in the consequences of an accident previously evaluated because there is no change to the status of any system required to mitigate the consequences of an accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because this change will reduce additional stresses on the RCS pressure boundary. Additionally, extending the required completion time for plant cooldown does not significantly increase the probability that significant additional RCS deterioration will occur prior to the plant being in Mode 5.

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LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change increases the limit for RCS total leakage (identified plus unidentified) from 10 gpm to 11 gpm. This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because the difference between a 10 gpm and an 11 gpm limit for total leakage (identified plus unidentified) is not significant considering that the limits are action levels that are indicative of potentially significant RCS boundary deterioration. These action levels are based on conservative engineering judgement and do not have intrinsic safety significance. Therefore, this change will not affect the timely identification and response to RCS leakage that is indicative of significant RCS pressure boundary deterioration.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the difference between a 10 gpm and an 11 gpm limit for total leakage (identified plus unidentified) is not significant considering that the limits are action levels that are indicative of potentially significant RCS boundary deterioration. These action levels are based on conservative engineering judgement and do not have intrinsic safety significance. Therefore, this change will not affect the timely identification and response to RCS leakage that is indicative of significant RCS pressure boundary deterioration.

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LESS RESTRICTIVE
("L.3" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change allows 4 hours to reduce identified leakage to within limits before a reactor shutdown is required and extends the time permitted to perform a plant shutdown and cooldown from 28 hours to 36 hours. This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because the allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. Additionally, these changes do not significantly increase the probability that significant additional RCS deterioration will occur prior to the plant being in Mode 5.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. Additionally, these changes do not significantly increase the probability that significant additional RCS deterioration will occur prior to the plant being in Mode 5.

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LESS RESTRICTIVE
("L.4" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.4.13 - RCS Operational LEAKAGE

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change provides an additional 4 hours to evaluate primary to secondary leakage when limits are exceeded before a reactor shutdown and cooldown is required and extends the time permitted to perform a plant shutdown and cooldown when primary to secondary leakage limits are exceeded from 28 hours to 36 hours. This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because the allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. These changes are acceptable because extending the required completion time for plant cooldown does not significantly increase the probability of significant additional SG tube deterioration prior to the plant being in Mode 5.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. These changes are acceptable because extending the required completion time for plant cooldown does not significantly increase the probability of significant additional SG tube deterioration prior to the plant being in Mode 5.

LESS RESTRICTIVE
("L.5" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change eliminates special reporting and restarting requirements related to steam generator tube leaking that were added to the CTS because of concerns regarding the condition of the Indian Point 3 steam generators in 1979. These requirements are eliminated because the IP3 steam generators were replaced in 1989. This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because adequate controls to ensure the integrity of the steam generators are included in ITS 3.4.13, Operational Leakage, and ITS 5.5.8, Steam Generator Tube Surveillance Program.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the special requirements are eliminated because the IP3 steam generators were replaced in 1989. Therefore, adequate controls to ensure the integrity of the steam generators are included in ITS 3.4.13, Operational Leakage, and ITS 5.5.8, Steam Generator Tube Surveillance Program.

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LESS RESTRICTIVE
("L.6" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change reduces the Frequency for evaluating RCS leakage from 5 days/week to once every 72 hours and adds an allowance that the SR needs to be performed only during steady state operation. This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because a 72 hour frequency is a reasonable interval to trend leakage and provide early indication of gradual RCS deterioration. Additionally, the 5 days/week Frequency can be reduced because this verification is not used for the prompt identification of rapid changes in RCS leakage rates and other methods that provide prompt and sensitive indication of significant increases in RCS leakage are available to the operators, including RCS Leakage detection instrumentation required by ITS LCO 3.4.15. Therefore, reducing the Frequency for evaluating RCS leakage using an inventory balance will not affect the timely identification and response to RCS leakage that is indicative of significant RCS pressure boundary deterioration.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because a 72 hour frequency is a reasonable interval to trend leakage and provide early indication of gradual RCS deterioration. Additionally, the 5 days/week Frequency can be reduced because this verification is not used for the prompt identification of rapid changes in RCS leakage rates and other methods that provide prompt and sensitive indication of significant increases in RCS leakage are available to the operators, including RCS Leakage detection instrumentation required by ITS LCO 3.4.15. Therefore, reducing the Frequency for evaluating RCS leakage using an inventory balance will not affect the timely identification and response to RCS leakage that is indicative of significant RCS pressure boundary deterioration.

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LESS RESTRICTIVE
("L.7" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change eliminates the requirement for periodic evaluation of RCS leakage using an RCS water inventory balance in Mode 3 and 4 until there is 12 hours of steady state operation. This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because this change recognizes that an RCS water inventory balance requires steady state operating conditions and near operating pressure to yield valid results. Therefore, this change allows entry into Modes 3 and 4 without performing an RCS water inventory balance. This change is acceptable because the RCS water inventory balance must be completed prior to entering Mode 2 which ensures that an assessment of leakage is performed prior to establishing conditions where RCS pressure boundary integrity is most important. This change will not affect the timely identification and response to RCS leakage that is indicative of significant RCS pressure boundary deterioration.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because this change recognizes that an RCS water inventory balance requires steady state operating conditions and near operating pressure to yield valid results. Therefore, this change allows entry into Modes 3 and 4 without performing an RCS water inventory balance. This change is acceptable because the RCS water inventory balance must be completed prior to entering Mode 2 which ensures that an assessment of leakage is performed prior to establishing conditions where RCS pressure boundary integrity is most important. This change will not affect the timely identification and response to RCS leakage that is indicative of significant RCS pressure boundary deterioration.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.4.13:
# "RCS Operational LEAKAGE"

## PART 5:

## NUREG-1431
## Annotated to show differences between NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.4.13

This ITS Specification is based on NUREG-1431 Specification No. **3.4.13**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| CEOG-006 | **054 R1** | OPERATIONAL LEAKAGE TS BASES CHANGED TO BE CONSISTENT WITH THE IDENTIFIED LEAKAGE DEFINITION | Approved by NRC | Incorporated | **T.1** |
| CEOG-013 | **061** | ADDED STATEMENT CLARIFYING THE INTENT OF THE RCS WATER INVENTORY BALANCE SURVEILLANCE | Approved by NRC | Incorporated | **T.2** |
| CEOG-046 | **138 R0** | ADDITION OF ACTION FOR INOPERABLE STEAM GENERATOR | NRC Rejects: TSTF to Revise | Not Incorporated | **N/A** |
| WOG-050 | **116 R1** | RCS INVENTORY BALANCE SR: STEADY STATE CLARIFICATION | NRC Review | Not Incorporated | **N/A** |

<CTS>

## 3.4 REACTOR COOLANT SYSTEM (RCS)

<3.1.F>

### 3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

<3.1.F.4> a. No pressure boundary LEAKAGE;

<3.1.F.2> <DOC L.2> b. 1 gpm unidentified LEAKAGE;

<3.1.F.3> <3.1.F.5> c. 10 gpm identified LEAKAGE;

<3.1.F.8> d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and

<3.1.F.8> e. ~~[500]~~ 432 gallons per day primary to secondary LEAKAGE through any one SG.

<DOC A.6> APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

<3.1.F.2> <3.1.F.5> <3.1.F.8> <DOC L.1> <DOC L.3> <DOC L.4>

<3.1.F.2> <3.1.F.5> <3.1.F.8> <DOC L.1> <DOC L.3> <DOC L.4> <3.1.F.4>

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE. | A.1 Reduce LEAKAGE to within limits. | 4 hours |
| B. Required Action and associated Completion Time of Condition A not met.<br>OR<br>Pressure boundary LEAKAGE exists. | B.1 Be in MODE 3.<br>AND<br>B.2 Be in MODE 5. | 6 hours<br><br>36 hours |



Typical

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| SR 3.4.13.1 | ------------------NOTE------------------ Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. ------------------------------------------ Insert: 3.4-34-01 ~~Perform~~ RCS water inventory balance. | -----NOTE------ Only required to be performed during steady state operation -------------- 72 hours |
| SR 3.4.13.2 | Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program. | In accordance with the Steam Generator Tube Surveillance Program |

<Doc L.6>
<Doc A.7>
<Doc L.7>
<T 4.1-3, Item 7>
T.2
<Doc A.5>

**INSERT: 3.4-34-01**

T.2

Verify RCS Operational leakage is within limits by performance of

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.13 RCS Operational LEAKAGE

BASES

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BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

mo-A 10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. ~~Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.~~ CLB.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

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(continued)

BASES (continued)

**APPLICABLE SAFETY ANALYSES**

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for ~~an~~ event (a) resulting in steam discharge to the atmosphere assumes ~~a 1 gpm~~ a range of primary to secondary LEAKAGE from 0.1 gpm to 10 gpm as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. ~~3~~ 2) analysis for SGTR assumes the contaminated secondary fluid is ~~only briefly~~ released via safety valves ~~and the majority is steamed to the condenser~~ and atmospheric dump valves. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes ~~1 gpm~~ a range of primary to secondary LEAKAGE ~~in one generator~~ as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 ~~or~~ and the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

**LCO**

RCS operational LEAKAGE shall be limited to:

a. <u>Pressure Boundary LEAKAGE</u>

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount ~~that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period.~~ and is consistent with the capability of the equipment required by LCO 3.4.15, RCS Leakage Detection Instrumentation. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of ~~identified~~ unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE, the leakage into closed systems or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system. T.1

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm (1440 gpd) through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary to Secondary LEAKAGE through Any One SG

The ~~500~~ 432 gallons per day (0.3 gpm) limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

Insert: B3.4-76-01

~~LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.~~

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses

(continued)

INSERT: B 3.4-76-01 CLB.1

Leakage past PIVs or other leakage into closed systems is that leakage that can be accounted for and contained by a system not directly connected to the atmosphere. Leakage past PIVs or other leakage into closed systems is not included in the limits for either identified or unidentified LEAKAGE but PIV leakage must be within the limits specified for PIVs in LCO 3.4.14, "RCS Pressure Isolation Valves (PIV)." Leakage past PIVs or other leakage into closed systems is quantified before being exempted from the limits for identified LEAKAGE.

BASES

ACTIONS

B.1 and B.2 (continued)

acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

blowdown

The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. Therefore, this SR is not required to be performed in MODES 3 and 4 until 12 hours of steady state operation near operating pressure have been established.

Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Insert: B3.4-77-01

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage

(continued)

INSERT: B 3.4-77-01

, measured leakage past PIVs, and other leakage into closed systems

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

detection in the prevention of accidents. A Note under the Frequency column states that this SR is required to be performed during steady state operation.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. ~~Regulatory Guide 1.45, May 1973.~~

2 ~~3~~. FSAR, Section ~~[15]~~ 14.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.4.13:
# "RCS Operational LEAKAGE"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

## JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
## ITS SECTION 3.4.13 - RCS Operational LEAKAGE

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 Pressure isolation valve leakage and other leakage into closed systems is not included in determining reactor coolant system operational leakage to meet the requirements of CTS 3.1.F, therefore it is not included in the allowable identified leakage in the ITS.

CLB.2 Reference to Regulatory Guide 1.45 is deleted from the Bases. IP3 is not committed to Regulatory Guide 1.45.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DB.2 The IP3 safety analyses for events resulting in steam discharge to the atmosphere assume a range of primary to secondary leakage from 0.1 to 10.0 gpm.

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.4.13 - RCS Operational LEAKAGE

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-54, Rev.1 (CEOG-06) which changes the Operational Leakage TS Bases to be consistent with the Identified Leakage definition. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.2 This change incorporates Generic Change TSTF-61, Rev.0 (CEOG-13) which clarifies the intent of the water inventory balance surveillance, and makes it consistent with the wording of the other surveillances in the specification and with other surveillances in NUREG-1431. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.4.14:
# "RCS Pressure Isolation Valve (PIV) Leakage"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except leakage limits for valves in the residual heat removal (RHR) flow path when in, or during the transition to or from, the RHR mode of operation.

ACTIONS

------------------------------------ NOTES ------------------------------------

1. Separate Condition entry is allowed for each flow path.

2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

-------------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more flow paths with leakage from one or more RCS PIVs not within limit. | -------------NOTE------------- Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.1 and be in the reactor coolant pressure boundary or the high pressure portion of the system. ------------------------------ | |
| | A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve. | 4 hours |
| | <u>AND</u> | |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. (continued) | A.2.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.<br><br><u>OR</u><br><br>A.2.2 Restore RCS PIV to within limits. | 72 hours<br><br><br><br>72 hours |
| B. Required Action and associated Completion Time for Condition A not met. | B.1 Be in MODE 3.<br><br><u>AND</u><br><br>B.2 Be in MODE 5. | 6 hours<br><br><br><br>36 hours |
| C. RHR System autoclosure interlock function inoperable. | ------------NOTE------------<br>RHR system flowpath may be unisolated under administrative controls if needed to meet requirements for an operating RCS loop.<br>----------------------------<br><br>C.1 Isolate the affected penetration by use of one closed manual or deactivated automatic valve. | <br><br><br><br><br><br>4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.14.1 | -------------------NOTES-------------------<br>1. Not required to be performed in MODES 3 and 4.<br><br>2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation.<br><br>3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided.<br>---------------------------------------------<br><br>Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig. | 24 months<br><br><u>AND</u><br><br>Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 12 months<br><br><u>AND</u><br><br>Within 24 hours following valve actuation due to automatic or manual action or flow through the valve |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.14.2 | Verify RHR System autoclosure interlock prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ 450 psig. | 24 months |
| SR 3.4.14.3 | Verify RHR System autoclosure interlock causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ 550 psig. | 24 months |

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety. This LCO establishes limits for Event V PIVs only. Event V PIVs are defined as two check valves in series at a low pressure/RCS interface whose failure may result in a LOCA that by-passes containment. Event V refers to the scenario described for this event in the WASH-1400 study (Refs. 4 and 9). The Event V PIVs are listed in FSAR, Section 6 (Ref. 6).

The PIV leakage limit applies to each individual valve. Leakage through PIVs into closed systems is not included in the limits for either identified or unidentified LEAKAGE in LCO 3.4.13, RCS Operational LEAKAGE. Leakage past PIVs into closed systems is that leakage which can be accounted for and contained by a system not directly connected to the atmosphere.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a

BASES

BACKGROUND (Continued)

significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are typically provided to isolate the RCS from the following connected systems:

a. Residual Heat Removal (RHR) System; and

b. Safety Injection System.

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

Residual Heat Removal System Valves 730 and 731 isolate the RHR System from the RCS and are separately interlocked with independent pressure control signals to prevent their being opened whenever the RCS pressure is greater than a designated setpoint (which is below the RHR System design pressure of 600 psig). This interlock also automatically closes the valve whenever the Reactor Coolant System pressure increases to a slightly higher setpoint. This interlock provides a diverse backup to administrative requirements to close the isolation valves when needed to prevent RHR system overpressurization. In addition to this interlock, the valve motor operators are mechanically sized such that there is insufficient torque to open the valve in the presence of a pressure differential greater than the RHR System design pressure. Finally, the RHR System is equipped with a pressure relief valve sized to relieve the flow of two charging pumps. Collectively, these features provide a diverse backup to administrative requirements to close the isolation valves when needed to prevent RHR system overpressurization.

BASES

## APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment.

The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

The RHR isolation valve autoclosure and interlock provides a diverse backup to administrative requirements to close the isolation valves when needed to prevent RHR system overpressurization.

RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36.

## LCO

This LCO establishes limits for Event V PIVs only. Event V PIVs are defined as two check valves in series at a low pressure/RCS interface whose failure may result in a LOCA that by-passes containment. Event V refers to the scenario described for this event in the WASH-1400 study (Refs. 4 and 9). The Event V PIVs are listed in FSAR, Section 6 (Ref. 6).

RCS PIV leakage is leakage into closed systems connected to the RCS. Leakage through PIVs into closed systems is not included in the limits for either identified or unidentified LEAKAGE in LCO

BASES

LCO (continued)

3.4.13, RCS Operational LEAKAGE. Leakage past PIVs into closed systems is that leakage which can be accounted for and contained by a system not directly connected to the atmosphere. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

The autoclosure interlock for RHR System Valves 730 and 731 must function to automatically close or prevent the opening of the RHR isolation valves whenever the RCS pressure is greater than the RHR System design pressure. The autoclosure interlock is considered OPERABLE when the isolation valves are closed and the motor operators de-energized if the interlock would function when power is restored to the motor operator.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

BASES

APPLICABILITY (continued)

In MODES 5 and 6, leakage limits and RHR autoclosure function are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB or the high pressure portion of the system.

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period. If use of a closed

ACTIONS

A.1 and A.2 (continued)

manual, deactivated automatic, or check valve to isolate leaking PIV renders a required system or component inoperable, then the Required Actions associated with the affected system or component are initiated when the valve is closed.

B.1 and B.2

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

The inoperability of the RHR autoclosure interlock renders the RHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the RHR autoclosure interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the autoclosure function.

A Note to Required Action C.1 specifies that the RHR system flowpath may be unisolated under administrative controls if needed to meet requirements for an operating RCS loop in LCO 3.4.5, RCS Loops - MODE 3, and LCO 3.4.6, RCS Loops - MODE 4. Additionally, an RHR loop may be considered OPERABLE but not in operation with one or both RHR isolation valves closed and deactivated if the valves can be opened as allowed by this Note in a reasonable time. This Note is needed because neither of the two RHR loops can be in operation when either RHR valve 730 or 731 is closed.

BASES

ACTIONS

C.1 (continued)

This allowance is acceptable because the interlock is intended to provide a diverse backup to administrative requirements to close the isolation valves when needed to prevent RHR system overpressurization. In addition to this interlock, the valve motor operators are mechanically sized such that there is insufficient torque to open the valve in the presence of a pressure differential greater than the RHR System design pressure and the RHR System is equipped with a pressure relief valve sized to relieve the flow of three charging pumps.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 24 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 12 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

SR 3.4.14.2 and SR 3.4.14.3

Verifying that the RHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond 125% of its design pressure of 600 psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < 450 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.2 and SR 3.4.14.3 (continued)

the RHR relief valves will not lift. The 24 month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. FSAR Section 6.2.
7. ASME, Boiler and Pressure Vessel Code, Section XI.
8. 10 CFR 50.55a(g).
9. Generic Letter 87-006, Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.14:
## "RCS Pressure Isolation Valve (PIV) Leakage"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| T 4.1-3(2) | 148 | 148 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-7 | 178 | 178 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-8 | 178 | 178 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-11 | 148 | 148 | No TSCRs | No TSCRs for this Page | N/A |

at ≥ 550 psig

at ≥ 450 psig

M.3

TABLE 4.1-3 (Sheet 2 of 2)

| | | | |
|---|---|---|---|
| SR 3.4.14.2<br>SR 3.4.14.3 | 13. RHR Valves 730 and 731 | Automatic isolation and interlock action | 24M |
| SEE ITS 3.4.11 | 14. PORV Block Valves | Operability through 1 complete cycle of full travel | Quarterly (see Note 1) |
| | 15. PORV Valves | Operability | 24M |
| SEE RELOCATED | 16. Reactor Vessel Head Vents | Operability | 24M |

~~24M - At least once per 24 months~~

SEE ITS 3.4.11 — Note 1. If the block valve is shut due to a leaking or inoperable PORV, Block Valve operability will be checked the next time the plant is in cold shutdown.

Add Condition C and associated Req Action — M.2

B. Component Tests

1. Pumps

SEE ITS 3.5.2 3.5.3 3.6.6 3.7.8

a. The safety injection pumps, residual heat removal pumps, containment spray pumps and the auxiliary component cooling water pumps shall be started at quarterly intervals. The recirculation pumps shall be started at least once per 24 months.

b. Acceptable levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least fifteen minutes.

2. Valves

SEE ITS 3.6.7

a. Each spray additive valve shall be cycled by operator action with the pumps shut down at least once per 24 months.

SEE ITS 3.5.1

b. The accumulator check valves shall be checked for operability at least once per 24 months.

LCO 3.4.14
SR 3.4.14.1

c. The following check valves PIVs shall be checked for gross leakage at least once per 24 months: A.7

| | | | |
|---|---|---|---|
| 857A & G | 857J | 857S & T | 897B |
| 857B | 857K | 857U & W | 897C |
| 857C | 857L | 895A | 897D |
| 857D | 857M | 895P | 838A |
| 857E | 857N | 895C | 838B |
| 857F | 857P | 895D | 838C |
| 857H | 857Q & R | 897A | 838D |

M.5

LA.1

Add Notes 1, 2 and 3 to SR 3.4.14.1 — L.1

Add Acceptance Criteria to SR 3.4.14.1 — M.4

Add LCO 3.4.14, Applicability — A.3

Add Condition A and B and associated Req. Acts. — M.1





Add Actions Notes 1 and 2 — A.4

SR 3.4.14.1

~~d.~~ In addition to 4.5.B.2.c, the following check valves shall be checked for gross leakage every time the plant is shut down and the reactor coolant system has been depressurized to 700 psig or less. This gross leakage test shall also be performed following valve maintenance, repair or other work which could unseat these check valves: L.2 M.6 A.5

| | | |
|---|---|---|
| 838A | 895A | 897A |
| 838B | 895B | 897B |
| 838C | 895C | 897C |
| 838D | 895D | 897D |

add second Freq for SR 3.4.14.1 L.2

Add third Freq for SR 3.4.14.1 M.6

LA.1

**Basis**

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally on standby during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes reactor trip, main feedwater isolation and containment isolation, and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is, therefore, to combine systems tests to be performed during plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action and verification made that the components receive the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[1]

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly (in accordance with Specification 4.1). The testing of the analog channel inputs is accomplished in the same manner as for the reactor protection system. The engineered safety features logic system is tested by means of test switches to simulate inputs from the analog channels. The test switches allow actuation of the master relay, while at the same time blocking the slave relays. Verification that the logic is accomplished is indicated by the matrix test light. The slave relay coil circuits are continuously verified by a built-in monitoring circuit. In addition, the active components (pumps and valves) are to be tested in accordance with the Indian Point 3 Inservice Testing Program. The pumps, specified in the Technical Specifications, are tested on a quarterly basis to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The exception to this quarterly test are the recirculation pumps which are tested during a refueling outage. The quarterly test interval is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent testing would result in increased wear over a long period of time.

A.1

For the eight flow distribution valves (856 A, C, D, E, F, H, J and K), verification of the valve mechanical stop adjustments is performed periodically to provide assurance that the high head safety injection flow distribution is in accordance with flow values assumed in the core cooling analysis.

Gross leakage testing of the reactor coolant system pressure isolation valves and the Low Pressure Injection(LPI)/residual heat removal(RHR) system valves reduces the probability of an inter-system LOCA[4]. These tests implement the requirements set forth in NRC generic letter dated February 23, 1980, regarding testing of LPI/RHR system check valves. This amendment provides a basis for the rescission of item A.5. of a Confirmatory Order issued by the Commission to Indian Point 3 in a letter dated, February 11, 1980. To satisfy ALARA requirements, gross leakage (>10 gpm) may be measured indirectly (i.e. using installed pressure and flow indications).

A.1

References

(1) FSAR Section 6.2

(2) FSAR Section 6.4

(3) FSAR Section 6.8

(4) WASH 1400

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.4.14:
# "RCS Pressure Isolation Valve (PIV) Leakage"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.4.14 - RCS Pressure Isolation Valve (PIV) Leakage

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 4.5.B.2.c and CTS 4.5.B.2.d establish requirements for testing RCS pressure isolation valves (PIVs) and CTS Table 4.1-3, Item 13, establishes requirements for the residual heat removal (RHR) automatic isolation and interlock function test; however, there is no specific statement of when these tests must be met. ITS LCO 3.4.14 maintains these requirements and includes a specific Applicability statement that the requirements must be met in Modes 1, 2, and 3, and in Mode 4 except for valves in the RHR flow path when in, or during the transition to or from the RHR mode of operation. The Applicability statement for ITS LCO 3.4.14 requires the PIVs to function as pressure isolation barriers whenever the plant is above 200°F. This is an administrative change with no impact on safety because it is a reasonable interpretation of the existing requirements.

A.4 CTS 4.5.B.2.c and CTS 4.5.B.2.d establish requirements for testing reactor coolant system PIVs and CTS Table 4.1-3, Item 13, establishes requirements for the residual heat removal (RHR) automatic isolation and interlock function test. ITS LCO 3.4.14 maintains these requirements and includes two new Notes that clarify the application of these requirements:

ITS LCO 3.4.14, Note 1, specifies that separate condition entry is allowed for each flow path. In conjunction with ITS Specification 1.3, "Completion Times," this Note provides direction consistent with the intent of the CTS for inoperable PIVs. Specifically, this note allows separate entry into an LCO 3.4.14 Condition for each PIV and separate tracking of Completion Times based on a PIV's time of entry into the Condition. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable PIV. Complying with the Required Actions for one inoperable PIV may allow continued operation, and subsequent inoperable PIVs are governed by separate Condition entry and application of associated Required Actions. This is an administrative change with no impact on safety because any differences between the existing requirements and ITS 3.4.14 are described and justified elsewhere in this discussion of changes.

ITS LCO 3.4.14, Note 2, requires entry into applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV or the Action taken in response to an inoperable PIV. This note requires an evaluation of affected systems if a PIV is inoperable because the leakage may have affected system Operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function. This is an administrative change with no impact on safety because it is a reasonable interpretation of the existing requirements.

A.5 CTS 4.5.B.2.d includes a requirement for selected PIVs that gross leakage testing must also be performed following valve maintenance, repair or other work that could unseat check valves. This requirement is not included in ITS because ITS SR 3.4.14.1 requires leakage testing within 24 hours following valve actuation due to automatic or manual action or flow through the valve. Additionally, ITS SR 3.0.1 requires that SRs are met whenever equipment is required to be Operable. The Bases for SR 3.0.1 include the clarification that upon completion of maintenance, appropriate post maintenance testing is required to declare equipment Operable. This includes ensuring applicable Surveillances are not failed. Therefore, CTS statements establishing requirements to verify SRs are met following maintenance can be deleted. This is an administrative change with no adverse impact on safety.

MORE RESTRICTIVE

M.1 CTS 4.5.B.2.c and CTS 4.5.B.2.d establish requirements for testing RCS pressure isolation valves (PIVs); however, no actions are specified for the failure to meet these requirements.

ITS 3.4.14, Required Actions A.1 and A.2, are added to address one or more flow paths with leakage from one or more PIVs not within limits. These Actions require isolating the high pressure portion of an affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve that meet required leakage limits within 4 hours and isolated with a second valve within 72 hours. If these Required Actions and associated Completion Times are not met, then ITS 3.4.14, Required Actions B.1 and B.2, require the plant be placed in Mode 3 within 6 hours and Mode 5 within 36 hours. The associated Bases provide the clarification that if use of a closed manual, deactivated automatic, or check valve to isolate leaking PIV renders a required system or component inoperable, then the Required Actions associated with the affected system or component are initiated when the valve is closed.

These changes are needed because isolating the high pressure portion of an affected system from the low pressure portion satisfies the safety function of the PIV (i.e., prevents overpressure failure of the low pressure portions of connecting systems). This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring a more conservative response than is currently required when PIV leakage is not within required limits. Therefore, this change has no significant adverse impact on safety.

M.2 CTS Table 4.1-3, Item 13, establishes requirements for the residual heat removal (RHR) automatic isolation and interlock test; however, no actions are specified for the failure to meet this requirement.

ITS 3.4.14, Required Action C.1, is added to address inoperability of the RHR autoclosure interlock and requires isolating the affected penetration by use of one closed manual or deactivated automatic valve within 4 hours. This change is needed because isolating the affected penetration accomplishes the purpose of the autoclosure function (i.e., prevents overpressure failure of the low pressure portions of the RHR system).

A Note to Required Action C.1 specifies that the RHR system flowpath may be unisolated under administrative controls if needed to meet requirements for an operating RCS loop in LCO 3.4.5, RCS Loops - MODE 3, and LCO 3.4.6, RCS Loops - MODE 4. Additionally, an RHR loop may be considered OPERABLE but not in operation with one or both RHR isolation valves closed and deactivated if the valves can be opened as allowed by this Note in a reasonable time. This Note is needed because neither of the two RHR loops can be in operation when either RHR valve 730 or 731 is closed. Administrative controls to ensure that RCS pressure is not increased above the RHR system design pressure will normally consist of a verification of RCS pressure every hour but may be more or less restrictive consistent with plant conditions and scheduled evolutions. This allowance is acceptable because, as described in the FSAR, this interlock is intended to provide a diverse backup to administrative requirements to close the isolation valves when needed to prevent RHR system overpressurization. In addition to this interlock, the valve motor operators are mechanically sized such that there is insufficient torque to open the valve in the presence of a pressure differential greater than the RHR System design pressure and the RHR System is equipped with a pressure relief valve sized to relieve the flow of two charging pumps.

This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring a more conservative response than is currently required when the RHR autoclosure function is not Operable. Therefore, this change has no significant adverse impact on safety.

M.3 CTS Table 4.1-3, Item 13, requires that the automatic isolation and interlock function for RHR valves 730 and 731 be verified every 24 months; however, acceptance criteria (allowable values) for the function are not specified in the CTS.

ITS SR 3.4.14.2 and SR 3.4.14.3 maintain the requirement to verify the automatic isolation and interlock function for RHR isolation valves every 24 months; however, the ITS includes the acceptance criteria that the autoclosure interlock prevents the valves from being opened with an RCS pressure signal ≥ 450 psig, and that the autoclosure interlock causes the valves to close automatically with RCS pressure signal ≥ 550 psig. This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification of the function and setpoints assumed in the design for the overpressure protection of the RHR system. Therefore, this change has no significant adverse impact on safety.

M.4 CTS 4.5.B.2.c and CTS 4.5.B.2.d require that PIVs (See ITS 3.4.14, DOC LA.1) be checked periodically for gross leakage; however, no acceptance criteria for leakage is included in the CTS.

ITS SR 3.4.14.1 maintains the requirement that PIVs be checked periodically for gross leakage; however, ITS SR 3.4.14.1 includes the acceptance criteria that leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig. This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring specific acceptance criteria for PIV leakage that is consistent with ASME requirements. Therefore, this change has no significant adverse impact on safety.

M.5 CTS 4.5.B.2.c requires that PIVs (See ITS 3.4.14, DOC LA.1) be checked for gross leakage every 24 months. Additionally, CTS 4.5.B.2.d requires that PIVs in the injection flow path be checked for leakage whenever the reactor is shutdown and depressurized to less than 700 psig.

ITS SR 3.4.14.1 maintains the requirement that PIVs be checked for gross leakage every 24 months; however, ITS SR 3.4.14.1 requires that all PIVs be checked for leakage whenever the unit has been in Mode 5 for 7 days or more and only if leakage testing has not been performed in the previous 12 months (See ITS 3.4.14, DOC L.2). This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring more frequent testing of selected PIVs. Therefore, this change has no significant adverse impact on safety.

M.6 CTS 4.5.B.2.c and CTS 4.5.B.2.d require that PIVs (See ITS 3.4.14, DOC LA.1) be checked periodically for gross leakage (See ITS 3.4.14, DOC M.5 and L.2).

ITS SR 3.4.14.1 maintains the requirement that PIVs be checked periodically for gross leakage; however, ITS SR 3.4.14.1 includes a new requirement that testing for gross leakage must be performed within 24 hours following any valve actuation due to automatic or manual action or after any flow through the valve. PIVs disturbed in the performance of this Surveillance must also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. This change is needed because it provides greater assurance that PIVs are properly re-seated after any operation. This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 4.5.B.2.c and CTS 4.5.B.2.d require that PIVs (See ITS 3.4.14, DOC LA.1) be checked periodically for gross leakage.

ITS SR 3.4.14.1 maintains the requirement that PIVs be checked periodically for gross leakage; however, ITS SR 3.4.14.1 includes three

Notes that provide needed relaxations from certain testing requirements as follows:

ITS SR 3.4.14.1, Note 1, specifies that PIV leak testing is not required to be performed in Modes 3 and 4 although the SR is required to be met. This Note is needed because entry into Modes 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note is acceptable because the SR must be met in Modes 3 and 4 although not performed and the SR must be performed prior to entering Modes 1 and 2. Therefore, there is a low probability that the change will prevent timely identification of a PIV with excessive leakage.

ITS SR 3.4.14.1, Note 2, specifies that PIV leak testing is not required to be performed on the PIVs located in the RHR flow path when in the shutdown cooling mode of operation. This change is needed and is acceptable because the exempted PIVs are open to maintain the RHR flowpath and the SR must be performed after RHR is secured. Therefore, there is a low probability that the change will prevent timely identification of a PIV with excessive leakage.

ITS SR 3.4.14.1, Note 3, specifies that PIVs actuated during the performance of SR 3.4.14.1 are not required to be tested more than once if a repetitive testing loop cannot be avoided. This change is needed and is acceptable because it recognizes that plant configuration may not support not disturbing a valve after it has been satisfactorily tested.

L.2 CTS 4.5.B.2.d requires that PIVs in the injection flow path (See ITS 3.4.14, DOC M.5) be checked for gross leakage whenever the reactor is shutdown and depressurized to less than 700 psig.

ITS SR 3.4.14.1 relaxes this requirement by limiting the leakage testing to prior to entering Mode 2 whenever the unit has been in Mode 5 for 7 days or more and only if leakage testing has not been performed in the previous 12 months. This change is needed so that PIV leakage testing is required only during significant plant shutdowns and only if the PIVs have not been tested recently. This change is acceptable because of the new requirement that PIVs must be tested for gross leakage within 24 hours following valve actuation due to automatic or manual action or flow through the valve (See ITS 3.4.14, DOC M.6). This change has no significant adverse impact on safety because the reduced Frequency for PIV leakage testing applies only to valves that have not been disturbed since the last satisfactory leakage test. Therefore, there is a low probability that the change will prevent timely identification of a PIV with excessive leakage.

REMOVED DETAIL

LA.1 CTS 4.5.B.2.c and CTS 4.5.B.2.d requires periodic leak tests of PIVs and includes a list of the applicable valves.

ITS SR 3.4.14.1 maintains the requirement to periodically leak test PIVs that are currently listed in the CTS but the list of PIVs governed by ITS SR 3.4.14.1 is relocated to the FSAR.

Maintaining the list of PIVs that must be tested as required by ITS SR 3.4.14.1 in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, are designed to assure that changes to the FSAR do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.14:
## "RCS Pressure Isolation Valve (PIV) Leakage"

## PART 4:

## No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

# NO SIGNIFICANT HAZARDS EVALUATION
## ITS SECTION 3.4.14 - RCS Pressure Isolation Valve (PIV) Leakage

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS SR 3.4.14.1 maintains the requirements in CTS 4.5.B.2.c and CTS 4.5.B.2.d that PIVs be checked periodically for gross leakage; however, ITS SR 3.4.14.1 includes three Notes that provide needed relaxations from certain testing requirements as follows:

   Note 1 specifies that PIV leak testing is not required to be performed in Modes 3 and 4 although the SR is required to be met. This Note is needed because entry into Modes 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note is acceptable because the SR must be met in Modes 3 and 4 although not performed and the SR must be performed prior to entering Modes 1 and 2.

   Note 2 specifies that PIV leak testing is not required to be performed on the PIVs located in the RHR flow path when in the shutdown cooling mode of operation. This Note is needed and is acceptable because the exempted PIVs are open to maintain the RHR flowpath and the SR must be performed after RHR is secured.

   Note 3 specifies that PIVs actuated during the performance of SR 3.4.14.1 are not required to be tested more than once if a repetitive testing loop cannot be avoided. This change is needed and is acceptable because it recognizes that plant configuration may not support not disturbing a valve after it has been satisfactorily tested.

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because PIV leakage SRs must be met at all times the PIVs are required to be Operable. These notes provide recognition that performance of the SRs must be deferred until plant conditions or plant configuration supports performance of the leakage tests.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because PIV leakage SRs must be met at all times the PIVs are required to be Operable. These notes provide recognition that performance of the SRs must be deferred until plant conditions or plant configuration supports performance of the leakage tests.

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LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS SR 3.4.14.1 relaxes the CTS requirement for leak testing PIVs when shutdown by limiting the leakage testing to prior to entering Mode 2 whenever the unit has been in Mode 5 for 7 days or more and only if leakage testing has not been performed in the previous 12 months. This change is needed so that PIV leakage testing is required only during significant plant shutdowns and only if the PIVs have not been tested recently. This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because of the new requirement that PIVs must be tested for gross leakage within 24 hours following valve actuation due to automatic or manual action or flow through the valve. Therefore, the reduced Frequency for PIV leakage testing applies only to valves that have not been disturbed since the last satisfactory leakage test and there is very little probability that the change will prevent timely identification of a PIV with excessive leakage.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because of the new requirement that PIVs must be tested for gross leakage within 24 hours following valve actuation due to automatic or manual action or flow through the valve. Therefore, the reduced Frequency for PIV leakage testing applies only to valves that have not been disturbed since the last satisfactory leakage test and there is very little probability that the change will prevent timely identification of a PIV with excessive leakage.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.14:
## "RCS Pressure Isolation Valve (PIV) Leakage"

## PART 5:

## NUREG-1431
## Annotated to show differences between
## NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.4.14

This ITS Specification is based on NUREG-1431 Specification No. **3.4.14**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| N/A | N/A | NO GENERIC CHANGES ARE POSTED AGAINST THIS SPECIFICATION. | Not Applicable | Not Applicable | N/A |

<CTS>

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

<4.5.B.2.c>
<T4.1-3, #13>

LCO 3.4.14 Leakage from each RCS PIV shall be within limit.

Insert: 3.4-35-01 (PA.2)

<Doc A.3>

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the residual heat removal (RHR) flow path when in, or during the transition to or from, the RHR mode of operation.

ACTIONS

<Doc A.4>

-----NOTES-----

1. Separate Condition entry is allowed for each flow path.
2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

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<Doc M.1>

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more flow paths with leakage from one or more RCS PIVs not within limit. | -----NOTE----- Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.1 and be in the reactor coolant pressure boundary ~~[~~or the high pressure portion of the system~~]~~. | (continued) |

INSERT: 3.4-35-01

for leakage limits for

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. (continued) | A.1 | Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve. | 4 hours |
| | AND | | |
| | A.2.1 | Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve. | 72 hours |
| | OR | | |
| | A.2.2 | Restore RCS PIV to within limits. | 72 hours |
| B. Required Action and associated Completion Time for Condition A not met. | B.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | B.2 | Be in MODE 5. | 36 hours |
| C. RHR System autoclosure interlock function inoperable. | C.1 | Isolate the affected penetration by use of one closed manual or deactivated automatic valve. | 4 hours |

<DOC M.1>
<DOC M.1>
<DOC M.1>
<DOC M.2>
X.1

Insert: 3.4-36-01

**INSERT: 3.4-36-01** X.1

----------Note----------
RHR system flowpath may be unisolated under administrative controls if needed to meet requirements for an operating RCS loop.
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SURVEILLANCE REQUIREMENTS

| | | SURVEILLANCE | FREQUENCY |
|---|---|---|---|
| | SR 3.4.14.1 | ------------------NOTES------------------ | |
| <Doc L.1> | | 1. Not required to be performed in MODES 3 and 4. | |
| <Doc L.1> | | 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. | |
| <Doc L.1> | | 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. | |
| | | ------------------------------------------ | |
| <4.5.B.2.d> <4.5.B.2.c> <Doc M.4> | | Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ [2215] psig and ≤ [2255] psig. | ~~In accordance with the Inservice Testing Program, and~~ ~~18~~ 24 months |
| | | | **AND** |
| <Doc L.2> <Doc M.5> | | | Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous ~~9~~ 12 months |
| | | | **AND** X.2 |
| | | | (continued) |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.4.14.1 (continued) | Within 24 hours following valve actuation due to automatic or manual action or flow through the valve |
| SR 3.4.14.2 ~~-------NOTE------- Not required to be met when the RHR System autoclosure interlock is disabled in accordance with SR 3.4.12.7. -------------~~ Verify RHR System autoclosure interlock prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ ~~[425]~~ 450 psig. | ~~[18]~~ 24 months |
| SR 3.4.14.3 ~~-------NOTE------- Not required to be met when the RHR System autoclosure interlock is disabled in accordance with SR 3.4.12.7. -------------~~ Verify RHR System autoclosure interlock causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ ~~[600]~~ 550 psig. | [18] 24 months |

<DOC M.6>

<Table 4.1-3, #13>

<Table 4.1-3, #13>

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

Insert: B3.4-79-02

Insert: B 3.4-79-01

The PIV leakage limit applies to each individual valve. ~~Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.~~ CLB.1

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

(continued)

INSERT: B 3.4-79-01

CLB.1

Leakage through PIVs into closed systems is not included in the limits for either identified or unidentified LEAKAGE in LCO 3.4.13, RCS Operational LEAKAGE. Leakage past PIVs into closed systems is that leakage which can be accounted for and contained by a system not directly connected to the atmosphere.

INSERT: B 3.4-79-02

CLB.1

This LCO establishes limits for Event V PIVs only. Event V PIVs are defined as two check valves in series at a low pressure/RCS interface whose failure may result in a LOCA that by-passes containment. Event V refers to the scenario described for this event in the WASH-1400 study (Refs 4 and 9). The Event V PIVs are listed in FSAR, Section 6 (Ref. 6).

BASES

BACKGROUND (continued)

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System; and

b. Safety Injection System~~, and~~

~~c. Chemical and Volume Control System.~~

The PIVs are listed in the FSAR, Section ~~[ ]~~ 6 (Ref. 6).

Insert B 3.4-80-01

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is ~~typically~~ designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

Insert: B 3.4-80-02

RCS PIV leakage satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

Insert: 3.4-80-04

RCS PIV leakage is ~~identified LEAKAGE~~ leakage into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases

Insert: B 3.4-80-03

(continued)

## INSERT: B 3.4-80-01

Residual Heat Removal System Valves 730 and 731 isolate the RHR System from the RCS and are separately interlocked with independent pressure control signals to prevent their being opened whenever the RCS pressure is greater than a designated setpoint (which is below the RHR System design pressure of 600 psig). This interlock also automatically closes the valve whenever the Reactor Coolant System pressure increases to a slightly higher setpoint. This interlock provides a diverse backup to administrative requirements to close the isolation valves when needed to prevent RHR system overpressurization. In addition to this interlock, the valve motor operators are mechanically sized such that there is insufficient torque to open the valve in the presence of a pressure differential greater than the RHR System design pressure. Finally, the RHR System is equipped with a pressure relief valve sized to relieve the flow of two charging pumps. Collectively, these features provide a diverse backup to administrative requirements to close the isolation valves when needed to prevent RHR system overpressurization.

## INSERT: B 3.4-80-02

The RHR isolation valve autoclosure and interlock provides a diverse backup to administrative requirements to close the isolation valves when needed to prevent RHR system overpressurization.

## INSERT: B 3.4-80-03

Leakage through PIVs into closed systems is not included in the limits for either identified or unidentified LEAKAGE in LCO 3.4.13, RCS Operational LEAKAGE. Leakage past PIVs into closed systems is that leakage which can be accounted for and contained by a system not directly connected to the atmosphere.

## INSERT: B 3.4-80-04

This LCO establishes limits for Event V PIVs only. Event V PIVs are defined as two check valves in series at a low pressure/RCS interface whose failure may result in a LOCA that by-passes containment. Event V refers to the scenario described for this event in the WASH-1400 study (Refs 4 and 9). The Event V PIVs are listed in FSAR, Section 6 (Ref. 6).

BASES

LCO (continued)

significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power. (PA.3)

Insert: B3.4-81-01

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO for leakage when in, or during the transition to or from, the RHR mode of operation. (PA.2)

In MODES 5 and ~~6~~, leakage limits and RHR autoclosure function are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment. (PA.3)

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have

(continued)

INSERT: B 3.4-81-01

PA3

The autoclosure interlock for RHR System Valves 730 and 731 must function to automatically close or prevent the opening of the RHR isolation valves whenever the RCS pressure is greater than the RHR System design pressure. The autoclosure interlock is OPERABLE when the isolation valves are closed and the motor operators de-energized if the interlock would function as soon as power is restored to the motor operator.

BASES

ACTIONS (continued)

degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB [for the high pressure portion of the system].

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

[Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

Insert: B3.4-82-01

~~or~~

~~The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period. (Reviewer Note: Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.)~~]

B.1 and B.2

If leakage cannot be reduced, [the system isolated,] or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3

(continued)

INSERT: B 3.4-82-01

If use of a closed manual, deactivated automatic, or check valve to isolate leaking PIV renders a required system or component inoperable, then the Required Actions associated with the affected system or component are initiated when the valve is closed.

BASES

ACTIONS

B.1 and B.2 (continued)

within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

The inoperability of the RHR autoclosure interlock renders the RHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the RHR autoclosure interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the autoclosure function.

Insert: B3.4-83-01

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every [18] 24 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The ~~[18 month]~~ 24 month Frequency is consistent with

(continued)

X.1

INSERT: B 3.4-83-01

A Note to Required Action C.1 specifies that the RHR system flowpath may be unisolated under administrative controls if needed to meet requirements for an operating RCS loop in LCO 3.4.5, RCS Loops - MODE 3, and LCO 3.4.6, RCS Loops - MODE 4. Additionally, an RHR loop may be considered OPERABLE but not in operation with one or both RHR isolation valves closed and deactivated if the valves can be opened as allowed by this Note in a reasonable time. This Note is needed because neither of the two RHR loops can be in operation when either RHR valve 730 or 731 is closed. Administrative controls to ensure that RCS pressure is not increased above the RHR system design pressure will normally consist of a verification of RCS pressure every hour but may be more or less restrictive consistent with plant conditions and scheduled evolutions. This allowance is acceptable because the interlock is intended to provide a diverse backup to administrative requirements to close the isolation valves when needed to prevent RHR system over-pressurization. In addition to this interlock, the valve motor operators are mechanically sized such that there is insufficient torque to open the valve in the presence of a pressure differential greater than the RHR System design pressure and the RHR System is equipped with a pressure relief valve sized to relieve the flow of two charging pumps.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in ~~MODE 5 for 7 days or more,~~ if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established. X.2

12

SR 3.4.14.2 and SR 3.4.14.3

Verifying that the RHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond 125% of its design pressure of [600] psig. The interlock setpoint that prevents the valves from being

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.2 and SR 3.4.14.3 (continued)

opened is set so the actual RCS pressure must be < 425 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The 18 month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

450

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~~These SRs are modified by Notes allowing the RHR autoclosure function to be disabled when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.~~

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, ~~Section V, GDC 55~~.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. ~~Document containing list of PIVs.~~ FSAR Section 6.
7. ASME, Boiler and Pressure Vessel Code, Section XI.
8. 10 CFR 50.55a(g).
9.

Insert: B 3.4-85-01

PA.1

INSERT: B 3.4-85-01

Generic Letter 87-006, Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.4.14:
# "RCS Pressure Isolation Valve (PIV) Leakage"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.4.14 - RCS Pressure Isolation Valve (PIV) Leakage

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 IP3 LCO 3.4.14 Bases and the IP3 definition of Leakage differ from NUREG-1431, Rev 1, in that pressure isolation valve (PIV) leakage into closed systems is not included in determining reactor coolant system operational identified leakage.(Leakage into closed systems is leakage that can be accounted for and contained by a system not directly connected to the atmosphere. Leakage past the pressurizer safety valve seats and leakage past the safety injection pressure isolation valves are examples of reactor coolant system leakage into closed systems.

This change to NUREG-1431, Rev 1, maintain the requirements found in CTS 3.1.F.2 and CTS 3.1.F.3 which establish limits for unidentified and total (unidentified and identified) RCS Leakage and specifies that these limits do not apply to controlled leakage sources such as the reactor coolant pump controlled leakage seals and leakage into closed systems.

This change, which increases the allowable RCS identified leakage by not counting leakage into closed systems, is acceptable because leakage limits are action levels that are indicative of potentially significant RCS boundary deterioration. Leakage past PIVs is measured separately and subject to their own specific leakage limits. Therefore, leakage past PIVs is indicative of significant PIV leakage for which Conditions and Required Actions are established. This is an administrative change with no significant adverse impact on safety because there is no change to the existing requirement.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PA.2 IP3 LCO 3.4.14, Applicability and the supporting Bases, differs from NUREG-1431, Rev 1, in that the phrase "for leakage limits for" was added to clarify that the exception to the LCO for RHR valves in Mode 4 applies to the RHR leakage limits and not the RHR autoclosure function. This change is needed because the NUREG-1431, Rev 1, IP3 LCO 3.4.14 LCO and Applicability statements are ambiguous regarding requirements for the RHR isolation autoclose feature. The IP3 change ensures that the autoclose function in Operable in Mode 4 when this function is most likely to be needed to protect the RHR system. This change improves clarity and ensures requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PA.3 IP3 LCO 3.4.14 Bases differ from NUREG-1431, Rev 1, in that the Background, Applicable Safety Analysis, LCO, and Applicability were revised to include the RHR autoclosure and interlock function. This change is needed because the RHR autoclosure and interlock function were not addressed in NUREG-1431, Rev. 1. This is an administrative change with no significant adverse impact on safety because there is no change to the NUREG-1431 requirements.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

X.1 IP3 LCO 3.4.14, Required Action C.1 and the supporting Bases, differs from NUREG-1431, Rev 1, in that a Note is added to specify that the RHR system flowpath may be unisolated under administrative controls if needed to meet requirements for an operating RCS loop in LCO 3.4.5, RCS Loops - MODE 3, and LCO 3.4.6, RCS Loops - MODE 4. Additionally, an RHR loop may be considered OPERABLE but not in operation with one or both RHR isolation valves closed and deactivated if the valves can be opened as allowed by this Note in a reasonable time. This Note is needed because neither of the two RHR loops can be in operation when either RHR valve 730 or 731 is closed. Administrative controls to ensure that RCS pressure is not increased above the RHR system design pressure will normally consist of a verification of RCS pressure every hour but may be more or less restrictive consistent with plant conditions and scheduled evolutions. This allowance is acceptable because the interlock is intended to provide a diverse backup to administrative requirements to close the isolation valves when needed to prevent RHR system overpressurization. In addition to this interlock, the valve motor operators are mechanically sized such that there is insufficient torque to open the valve in the presence of a pressure differential greater than the RHR System design pressure and the RHR System is equipped with a pressure relief valve sized to relieve the flow of two charging pumps.

This change is unique to IP3 because the IP3 design has both RHR loops protected by a single set of isolation valves and neither of the two RHR loops can be in operation when either RHR valve 730 or 731 is closed.

X.2 IP3 SR 3.4.14.1 and the supporting Bases differ from NUREG-1431, Rev 1, in that the limit on the conditional Frequency was changed from 9 months to 12 months. The conditional Frequency is intended to approximate the mid point in a normal refueling cycle. Therefore, the IP3 normal Frequency of 24 months for SR 3.4.14.1 supports a conditional Frequency of 12 months. This is an administrative change with no significant adverse impact on safety.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.15:
## "RCS Leakage Detection Instrumentation"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump discharge flow monitor;

b. One containment atmosphere radioactivity monitor (gaseous or particulate); and

c. One containment fan cooler unit condensate measuring system.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.4 is not applicable.
----------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Required containment sump flow monitor inoperable. | A.1 Perform SR 3.4.13.1.<br><br>AND<br><br>A.2 Restore required containment sump monitor to OPERABLE status. | Once per 24 hours<br><br><br><br>30 days |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| B. Required containment atmosphere radioactivity monitor inoperable. | B.1.1 | Analyze grab samples of the containment atmosphere. | Once per 24 hours |
| | <u>OR</u> | | |
| | B.1.2 | Perform SR 3.4.13.1. | Once per 24 hours |
| | <u>AND</u> | | |
| | B.2.1 | Restore required containment atmosphere radioactivity monitor to OPERABLE status. | 30 days |
| | <u>OR</u> | | |
| | B.2.2 | Verify containment fan cooler unit condensate measuring system is OPERABLE. | 30 days |
| C. Required containment fan cooler unit condensate measuring system inoperable. | C.1 | Perform SR 3.4.15.1. | Once per 8 hours |
| | <u>OR</u> | | |
| | C.2 | Perform SR 3.4.13.1. | Once per 24 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| D. Required containment atmosphere radioactivity monitor inoperable. | D.1 | Restore required containment atmosphere radioactivity monitor to OPERABLE status. | 30 days |
| AND | OR | | |
| Required containment fan cooler unit condensate measuring system inoperable. | D.2 | Restore required containment fan cooler unit condensate measuring system to OPERABLE status. | 30 days |
| E. Required Action and associated Completion Time not met. | E.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | E.2 | Be in MODE 5. | 36 hours |
| F. All required monitors inoperable. | F.1 | Enter LCO 3.0.3. | Immediately |

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| SR 3.4.15.1 | Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitor. | 12 hours |
| SR 3.4.15.2 | Perform COT of the required containment atmosphere radioactivity monitor. | 92 days |
| SR 3.4.15.3 | Perform CHANNEL CALIBRATION of the required containment sump flow monitor. | 24 months |
| SR 3.4.15.4 | Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor. | 24 months |
| SR 3.4.15.5 | Perform CHANNEL CALIBRATION of the required containment fan cooler unit condensate measuring system. | 24 months |

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE and containment fan cooler unit condensate measuring system are instrumented to alarm for increases of 0.5 to 1.0 gpm. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of $10^{-11}$ $\mu$Ci/cc radioactivity for particulate monitoring and of $10^{-7}$ $\mu$Ci/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate (R-11) and gaseous activities (R-12) because of their sensitivities and rapid responses to RCS LEAKAGE.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity

BASES

BACKGROUND (Continued)

levels of the containment atmosphere as an indicator of potential RCS LEAKAGE. A 1°F increase in dew point is well within the sensitivity range of available instruments.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump and condensate flow from fan cooler unit condensate measuring system. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. 2).

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public.

BASES

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APPLICABLE SAFETY ANALYSES (continued)

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36.

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LCO

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump flow monitor, in combination with a gaseous or particulate radioactivity monitor and a containment fan cooler unit condensate measuring system, provides an acceptable minimum. The condensate measuring system associated with any one of the fan cooler unit satisfies the requirement for a fan cooler unit condensate measuring system.

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APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

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ACTIONS

The Actions are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the required monitors are inoperable.

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BASES

ACTIONS (continued)

This allowance is provided because other instrumentation is available to monitor for RCS leakage.

A.1 and A.2

With the required containment sump flow monitor inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitor or containment fan cooler unit will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage.

Restoration of the required sump flow monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

B.1.1, B.1.2, B.2.1 and B.2.2

With both gaseous and particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere radioactivity monitors. Alternatively, continued operation is allowed if the air cooler unit condensate measuring system is OPERABLE, provided grab samples are taken or water inventory balance performed every 24 hours.

The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

BASES

ACTIONS (continued)

C.1 and C.2

With the required containment fan cooler unit condensate measuring system inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or a water inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment fan cooler unit condensate measuring system to OPERABLE status.

The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE.

D.1 and D.2

With the required containment atmosphere radioactivity monitor and the required containment fan cooler unit condensate measuring system inoperable, the only means of detecting leakage is the containment sump flow monitor. This Condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable required monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

E.1 and E.2

If a Required Action of Condition A, B, C, or D cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

F.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.15.3, SR 3.4.15.4 and SR 3.4.15.5

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. FSAR, Section 6.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.4.15:
# "RCS Leakage Detection Instrumentation"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

### Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.1-32 | 121 | 121 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-33 | 121 | 121 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-34 | 121 | 121 | No TSCRs | No TSCRs for this Page | N/A |
| T 4.1-1(2) | 169 | 169 | No TSCRs | No TSCRs for this Page | N/A |

A.1 A.2

Add Note: LCO 3.0.4 not applicable A.3

SEE ITS 3.4.13

9. If leakage from two or more tubes in the steam generators in any 20-day period is observed or determined, the reactor shall be brought to the hot shutdown condition within four hours and the cold shutdown condition within an additional twenty-four hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation. If two steam generator tube leaks attributable to the tube denting phenomena are observed after the reactor is in cold shutdown Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

SEE ITS 5.5.8

LCO 3.4.15 Applicability

Add LCO 3.4.15

Req. Act B.1, B.2

~~10.~~ When the reactor is critical and above 2% power, ~~two reactor coolant leak detection systems of different principles~~ capable of detecting leakage into containment shall be in operation, with one of the two systems sensitive to radioactivity. ~~The system sensitive to radioactivity may be out-of-service for 48 hours, provided two other systems are available.~~ M.2 M.1 L.1

~~**Basis:**~~

~~Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.~~ A.1

~~Although some leak rates on the order of GPM may be tolerable from a dose point of view, it must be recognized that small leaks through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.~~

~~The distinction between identified and unidentified leakage in the specification is made because once the leakage source is identified, the seriousness can be easily evaluated. The strict limit of 1 gallon per minute for unidentified leakage is adopted because in the worst case the leakage source may increase with time or the coolant may impinge on or accumulate in a critical component.~~

Add Condition A and associated Req Act M.3

Add Condition C and associated Req Act M.4

Add Conditions E and F and associated Req. Act M.7



Add Condition D and associated Req. Act. M.5

A.1

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Watch Force. Under these conditions, an allowable primary system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also well within the capacity of one charging pump and makeup would be available even under the loss of off-site power condition.

Controlled sources of reactor coolant system leakage are sources which are designed to leak at a controlled rate. For example, the reactor coolant pump seals are controlled leakage sources. Leakage through a valve packing or a closed valve is not considered as controlled leakage. Leakage into closed systems is that leakage which can be accounted for and contained by a system not directly connected to the atmosphere. Leakage past the pressurizer safety valve seats and steam generator tube leakage are examples of reactor coolant system leakage into closed systems.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor (R-11).

b. The containment radiogas monitor (R-12).

c. The containment humidity detectors.

d. A leakage detection system which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units.

The most sensitive and rapid method for detecting small amounts of Reactor Coolant System leakage is the monitoring of the containment airborne radioactivity. Containment gaseous and particulate activity is continuously, automatically monitored. The leakage rate can be determined by the relationship of the airborne activity to the reactor coolant activity.

Measurement of the leakage rate to the containment atmosphere is also possible through humidity detection and condensation collection and measurement. However, it is expected that the containment activity method will give the initial indication of coolant leakage. The other methods will be employed primarily to confirm that leakage exists, to indicate the location of the leakage sources, and to measure the leakage rate.

As described above, the four reactor coolant leak detection systems are based on three different principles, i.e., activity, humidity and condensate flow measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the containment.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory control.

Four hours is allowed from the time of leakage detection to identify the leakage source and to measure the leakage rate. This time period is required since identification and quantification of leakage sources of less than ten gallons per minute require a careful gathering and evaluation of data and/or a visual inspection of the reactor coolant system.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by the limitation of steam generator leakage between the primary coolant system and the secondary coolant system. Cracks having a primary-to-secondary leakage less than 500 gallons per day during operation will have an adequate margin of safaty against failure due to loads imposed by design basis accidents. The 500 gallon per day per steam generator limit is also consistent with the assumptions used to develop the Technical Specification limit on secondary coolant activity. Operating plants have demonstrated that primary-to-secondary leakage as low as 0.1 gpm will be detected. Leakage in excess of 432 gallons per steam generator or 1 gpm total for all four steam generators will require plant shutdown and an unscheduled eddy current inspection, during which the leaking tubes will be located and plugged.

<u>References</u>

FSAR Sections 11.2.3 and 14.2.4

A.1

Amendment No. ~~37~~, ~~34~~, 121

Add SR 3.4.15.5

Add SR 3.4.15.3

M.6

TABLE 4.1-1 (Sheet 2 of 6)

| Channel Description | Check | Calibrate | Test | Remarks |
|---|---|---|---|---|
| 8. 6.9 KV Voltage | N.A. | 18M | Q | Reactor protection circuits only |
| 6.9 KV Frequency | N.A. | 24M | Q | Reactor protection circuits only |
| 9. Analog Rod Position | S | 24M | M | |
| 10. Steam Generator Level | S | 24M | Q | |
| 11. Residual Heat Removal Pump Flow | N.A. | 24M | N.A. | |
| 12. Boric Acid Tank Level | S | 24M | N.A. | Bubbler tube rodded during calibration |
| 13. Refueling Water Storage Tank Level | | | | |
| a. Transmitter | W | 18M | N.A. | Low level alarm |
| b. Indicating Switch | W | 6M | N.A. | Low level alarm |
| 14a. Containment Pressure - narrow range | S | 24M | Q | High and High-High |
| 14b. Containment Pressure - wide range | M | 18M | N.A. | |
| 15. Process and Area Radiation Monitoring: | | | | |
| a. Fuel Storage Building Area Radiation Monitor (R-5) | D | 24M | Q | |
| b. Vapor Containment Process Radiation Monitors (R-11 and R-12) | ~~D~~ 12 hours | 24M | Q | |
| c. Vapor Containment High Radiation Monitors (R-25 and R-26) | D | 24M | Q | |
| d. Wide Range Plant Vent Gas Process Radiation Monitor (R-27) | D | 24M | Q | |

SEE CTS MASTER MARKUP

SR 3.4.15.1
SR 3.4.15.2
SR 3.4.15.4

SEE CTS MASTER MARKUP

M.8

Amendment No. 8, 38, 63, 68, 74, 93, 107, 125, 137, 140, 144, 148, 150, 154, 169

SR 3.4.15.1 SR 3.4.15.4 SR 3.4.15.2

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.4.15:
# "RCS Leakage Detection Instrumentation"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.4.15 - RCS Leakage Detection Instrumentation

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 ITS LCO 3.4.15, Actions Note, is added to provide an allowance that ITS LCO 3.0.4 is not applicable to RCS Leakage Detection Instrumentation. This allowance is needed because it permits entry into Modes 1, 2, 3 and 4 if one or more RCS Leakage Detection Systems are inoperable as long as the Required Actions and associated Completion Times are met for any applicable Condition. This change is acceptable because ITS

LCO 3.4.15 Required Actions permit operation to continue anywhere from 30 days to indefinitely when one or more RCS Leakage Detection Systems are inoperable as long as the compensatory Required Actions are performed. This is an administrative change with no impact on safety because there is no equivalent to LCO 3.0.4 in the CTS; therefore, providing an exception results in no changes to the existing requirements. The justification for adding LCO 3.0.4 is addressed in Discussion of Changes for ITS Section 1.0.

MORE RESTRICTIVE

M.1 CTS 3.1.F.10 requires operation of 2 RCS leak detection systems of different principles capable of detecting leakage into containment and requires that one of the required systems must be sensitive to radioactivity. The systems available to meet these requirements are the containment air particulate and radiogas monitors, the containment humidity detectors, and the containment fan cooler unit condensate measurement system.

ITS LCO 3.4.15 requires the 3 RCS leak detection systems and specifically requires the following: (a) one containment sump discharge flow monitor; (b) one containment atmosphere radioactivity monitor (gaseous or particulate); and, (c) one containment fan cooler unit condensate measuring system. Therefore, this change increases the required number of RCS leak detection systems from 2 to 3 with a specific requirement for the containment sump discharge flow monitor. Additionally, this change does not permit the containment humidity detectors to be used to satisfy LCO requirements.

This fist change is needed because the NRC considered the containment sump discharge flow monitor available for leakage detection in the approval of NYPA's application of "leak-before-break" technology as an alternative to providing protective devices against the dynamic loads resulting from postulated ruptures of the primary coolant loops (See NRC Letter, S. Varga to J. Brons, dated 3/10/86). The second change is needed because humidity detection is considered an indirect alarm or indication and is useful only to alert operators to a potential problem that warrants closer monitoring of other RCS leakage detection systems.

These changes are acceptable because they do not introduce any operation that is un-analyzed while requiring more diversity in the RCS leakage detection systems required to be Operable. Therefore, this change has no significant adverse impact on safety.

M.2 CTS 3.1.F.10 requires that the RCS leakage detection instrumentation Operable whenever the reactor is critical and above 2% power. ITS LCO 3.4.15, Applicability, requires that the RCS leakage detection instrumentation Operable in Modes 1, 2, 3, and 4. This change is needed to require monitoring whenever the potential exists for significant RCS leakage and crack propagation. This change also enhances the potential for early detection of RCS leakage indicative of pressure boundary deterioration prior to establishing high pressure and temperature conditions. This change is acceptable because it does not introduce any operation that is un-analyzed while requiring RCS leakage detection systems Operable whenever the potential for significant RCS leakage and crack propagation exist. Therefore, this change has no significant adverse impact on safety.

M.3 CTS 3.1.F.10 requires operation of 2 RCS leak detection systems of different principles and that one of the required systems is sensitive to radioactivity; however, CTS 3.1.F.10 includes Required Actions only for the inoperability of the system sensitive to radioactivity.

ITS LCO 3.4.15 requires Operability of 3 specific RCS leak detection systems (See ITS 3.4.15, DOC M.1) and one of the required systems must be the containment sump discharge flow monitor. In conjunction with this change, Required Actions A.1 and A.2, are added to address the Condition when the required sump flow monitor is inoperable. Specifically, Required Actions A.1 and A.2 require performance of an RCS water inventory balance every 24 hours in accordance with ITS SR 3.4.13.1 (See ITS 3.4.13, DOC L.6) as compensatory action whenever a required sump flow monitor is inoperable. Additionally, the sump flow monitoring capability must be restored within 30 days.

This change is needed to establish Conditions, Required Actions and

Completion Times for an inoperable containment sump flow monitoring because the NRC considered the containment sump discharge flow as a method available for leakage detection in their approval of NYPA's application of "leak-before-break" technology (See ITS 3.4.15, DOC M.1). This change is acceptable because it does not introduce any operation that is un-analyzed while requiring compensatory measures and restoration of an inoperable required RCS leakage detection system. Therefore, this change has no significant adverse impact on safety.

M.4 CTS 3.1.F.10 requires operation of 2 RCS leak detection systems of different principles and that one of the required systems is sensitive to radioactivity; however, CTS 3.1.F.10 includes Required Actions for the inoperability of the system sensitive to radioactivity only.

ITS LCO 3.4.15 requires Operability of 3 specific RCS leak detection systems (See ITS 3.4.15, DOC M.1) and one of the required systems must be an fan cooler unit condensate measuring system. In conjunction with this change, Required Actions C.1 and C.2 are added to address the Condition of the required fan cooler unit condensate measuring system inoperable. Specifically, Required Actions C.1 and C.2 provide two options for compensatory actions when the required fan cooler unit condensate measuring system is inoperable: (a) performing an RCS water inventory balance accordance with ITS SR 3.4.13.1 (See ITS 3.4.13, DOC L.6) once per 24 hours; or, (b) performing a Channel Check of the required containment atmosphere radioactivity monitor.

This change is needed to establish Conditions, Required Actions and Completion Times for an inoperable fan cooler unit condensate measuring system. This change is acceptable because it does not introduce any operation that is un-analyzed while requiring compensatory measures and restoration for an inoperable RCS leakage detection system. Therefore, this change has no significant adverse impact on safety.

M.5 CTS 3.1.F.10 requires operation of 2 RCS leak detection systems of different principles and that one of the required systems is sensitive to radioactivity; however, CTS 3.1.F.10 includes Required Actions for the inoperability of the system sensitive to radioactivity only.

ITS LCO 3.4.15, Condition D and associated Required Actions, is added to address the concurrent inoperability of the containment atmosphere radioactivity monitor and the containment fan cooler unit condensate measuring system. In this condition, one of the required systems must be restored within 30 days.

This change is needed because the containment sump discharge flow monitor is the only means of detecting RCS leakage if the containment atmosphere radioactivity monitor and the containment fan cooler unit condensate measurement system are both inoperable. In this situation, the required diverse means of leakage detection is not available. The Required Action is to restore either of the inoperable required monitors to Operable status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time is acceptable because the compensatory actions for each of the two inoperable systems will be performed in the interim.

This change is acceptable because it does not introduce any operation that is un-analyzed while requiring compensatory measures and eventual restoration for an inoperable required RCS leakage detection system. Therefore, this change has no significant adverse impact on safety.

M.6 CTS 3.1.F.10 requires 2 RCS leak detection systems and the systems available to meet these requirements include the containment fan cooler unit condensate measurement system; however, CTS does not include a requirement for the periodic Channel Calibration of the containment fan cooler unit condensate measuring system.

ITS SR 3.4.15.5 is added to require Channel Calibration of the containment fan cooler unit condensate measuring system every 24 months. ITS SR 3.4.15.3 is added to require Channel Calibration of the containment flow monitoring system every 24 months. These changes are needed to require periodic verification of the Operability of equipment required by ITS LCO 3.4.15. The 24 month Frequency is based on the need to perform this SR during a refueling outage and is consistent with the demonstrated reliability of the equipment. Therefore, this change has no significant adverse impact on safety.

M.7 ITS LCO 3.4.15, Condition E and associated Required Actions, addresses the situation when Required Actions or Completion Times for compensatory actions or restoration for inoperable RCS leakage detection systems cannot be met. In this situation, the plant must be brought to a Mode in which the requirement does not apply. To achieve this status, the plant must be brought to at least Mode 3 within 6 hours and to Mode 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

ITS LCO 3.4.15, Conditions F and associated Required Action, addresses the situation when all required RCS leakage detection systems are inoperable. In this situation, no automatic means of monitoring leakage are available, and an immediate plant shutdown is required in accordance with LCO 3.0.3.

These changes are acceptable because they do not introduce any operation that is un-analyzed while requiring a plant shutdown if requirements for RCS leakage detection systems cannot be met. Therefore, this change has no significant adverse impact on safety.

M.8 CTS Table 4.1-1, Item 15.b, requires a daily Channel Check of the vapor containment process radiation monitors (R-11 and R-12).

ITS SR 3.4.15.1 maintains this requirement for a Channel Check of these instruments; however, the Frequency is increased to every 12 hours. This change is needed because the Channel Check gives reasonable confidence that the channel is operating properly. Additionally, the Frequency of 12 hours is based on industry experience with instrument reliability and the need to detect off normal conditions in a timely manner. This change is acceptable because it does not introduce any operation that is un-analyzed while requiring more timely verification of instrument performance and more time identification of off normal conditions. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.1.F.10 allows the required leakage detection system sensitive to radioactivity may be out-of-service for 48 hours, provided two other systems are available.

ITS 3.4.15, Required Actions B.1 and B.2, allow the required containment atmosphere radioactivity monitor (gaseous or particulate) to be inoperable for 30 days if grab samples of the containment atmosphere are analyzed once per 24 hours or RCS operational leakage is verified to be within limits by performance of an RCS water inventory balance once per 24 hours. Alternatively, continued operation is also allowed if the containment fan cooler unit condensate measuring system is verified to be operable every 30 days, and either grab samples are taken every 24 hours or RCS operational leakage is verified to be within limits by performance of an RCS water inventory balance every 24 hours. This change is acceptable because the 24 hour interval to grab and analyze a containment air sample or perform a water inventory balance provides periodic information that is adequate to detect leakage, and the 30 day interval recognizes that at least one other form of leakage detection is available. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.4.15:
# "RCS Leakage Detection Instrumentation"

## PART 4:

## No Significant Hazards Considerations
## for
## Changes between CTS and ITS
## that are
## Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.4.15 - RCS Leakage Detection Instrumentation

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   CTS 3.1.F.10 states that the leakage detection system sensitive to radioactivity may be out-of-service for 48 hours, provided two other systems are available. ITS 3.4.15, Required Action B.1, revises the CTS requirement to allow the required containment atmosphere radioactivity monitor (gaseous or particulate) to be inoperable for 30 days if grab samples of the containment atmosphere are analyzed once per 24 hours or RCS operational leakage is verified to be within limits by performance of an RCS water inventory balance once per 24 hours. Alternatively, continued operation is also allowed if the containment fan cooler unit condensate measuring system is verified to be operable every 30 days, and either grab samples are taken every 24 hours or RCS operational leakage is verified to be within limits by performance of an RCS water inventory balance every 24 hours.

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because design, material, and construction standards applicable to the reactor coolant system are not being modified, and overall system performance is not affected. The change in the allowed outage time does not change any assumptions made in the accident analysis in evaluating radiological consequences. The 24 hour interval to grab and analyze a containment air sample or perform a water inventory balance provides periodic information that is adequate to detect leakage, and the 30 day interval recognizes that at least one other form of leakage detection is available.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because design, material, and construction standards applicable to the reactor coolant system are not being modified, and overall system performance is not affected. The change in the allowed outage time does not change any assumptions made in the accident analysis in evaluating radiological consequences. The 24 hour interval to grab and analyze a containment air sample or perform a water inventory balance provides periodic information that is adequate to detect leakage, and the 30 day interval recognizes that at least one other form of leakage detection is available.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.4.15:
# "RCS Leakage Detection Instrumentation"

## PART 5:

## NUREG-1431
## Annotated to show differences between NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.4.15

This ITS Specification is based on NUREG-1431 Specification No. **3.4.15**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| CEOG-012 | **060** | MAKE LCO 3.0.4 APPLICABLE TO ALL ACTIONS OF TS 3.4.15 | Approved by NRC | Incorporated | **T.1** |
| WOG-050 | **116 R1** | RCS INVENTORY BALANCE SR: STEADY STATE CLARIFICATION | NRC Review | Not Incorporated | **N/A** |

## 3.4 REACTOR COOLANT SYSTEM (RCS)

CTS

### 3.4.15 RCS Leakage Detection Instrumentation

<3.1.F10>

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

<DOC M.1>

a. One containment sump ~~(level or)~~ discharge flow~~(s)~~ monitor; DB.1

b. One containment atmosphere radioactivity monitor (gaseous or particulate); ~~,~~and

c. One containment ~~air cooler condensate flow rate monitor(s)~~. DB.1

Insert: 3.4-39-01

<DOC M.2> APPLICABILITY: MODES 1, 2, 3, and 4.

<DOC A.3> ACTIONS

<DOC M.3> T.1

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Required containment sump flow monitor inoperable. | ------------NOTE------------<br>LCO 3.0.4 is not applicable.<br>---------------------------- | |
| | A.1 Perform SR 3.4.13.1. | Once per 24 hours |
| | AND | |
| | A.2 Restore required containment sump monitor to OPERABLE status. | 30 days |

(continued)



Typical

INSERT: 3.4-39-01

fan cooler unit condensate measuring system

ACTIONS (continued)

<3.1.F10>
<DOC L.1>

T.1

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| B. Required containment atmosphere radioactivity monitor inoperable. | ~~NOTE~~ ~~LCO 3.0.4 is not applicable.~~ | | |
| | B.1.1 | Analyze grab samples of the containment atmosphere. | Once per 24 hours |
| | OR | | |
| | B.1.2 | Perform SR 3.4.13.1. | Once per 24 hours |
| | AND | | |
| | B.2.1 | Restore required containment atmosphere radioactivity monitor to OPERABLE status. | 30 days |
| | OR | | |
| | B.2.2 | Verify containment ~~air cooler condensate flow rate monitor~~ is OPERABLE. | 30 days |
| C. Required containment ~~air cooler condensate flow rate monitor~~ inoperable. | C.1 | Perform SR 3.4.15.1. | Once per 8 hours |
| | OR | | |
| | C.2 | Perform SR 3.4.13.1. | Once per 24 hours |

Insert: 3.4-40-01

DB.1

<DOC M.4>

(continued)

INSERT: 3.4-40-01

fan cooler unit condensate measuring system

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| ⟨DOC M.5⟩ D. Required containment atmosphere radioactivity monitor inoperable. <br> AND <br> Required containment air ~~cooler condensate flow rate monitor~~ inoperable. <br> Insert: 3.4-41-01 | D.1 | Restore required containment atmosphere radioactivity monitor to OPERABLE status. | 30 days |
| | OR | | |
| | D.2 | Restore required containment ~~air cooler condensate flow rate monitor~~ to OPERABLE status. | 30 days ⟨DB.1⟩ |
| ⟨DOC M.7⟩ E. Required Action and associated Completion Time not met. | E.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | E.2 | Be in MODE 5. | 36 hours |
| ⟨DOC M.7⟩ F. All required monitors inoperable. | F.1 | Enter LCO 3.0.3. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | | FREQUENCY |
|---|---|---|---|
| ⟨T4.1-1, Item 15.b⟩ ⟨DOC M.8⟩ SR | 3.4.15.1 | Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitor. | 12 hours |

(continued)

INSERT: 3.4-41-01

fan cooler unit condensate measuring system

SURVEILLANCE REQUIREMENTS (continued)

| | SURVEILLANCE | FREQUENCY | |
|---|---|---|---|
| ⟨T4.1-1, Item 15.b⟩ | SR 3.4.15.2 Perform COT of the required containment atmosphere radioactivity monitor. | 92 days | |
| ⟨DOC M.6⟩ | SR 3.4.15.3 Perform CHANNEL CALIBRATION of the required containment sump flow monitor. | ~~[18]~~ 24 months | DB.1 |
| ⟨T4.1-1, Item 15.b⟩ | SR 3.4.15.4 Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor. | ~~[18]~~ 24 months | |
| ⟨DOC M.6⟩ | SR 3.4.15.5 Perform CHANNEL CALIBRATION of the required containment ~~air cooler condensate flow rate monitor~~ Insert: 34-42-01 | ~~[18]~~ 24 months | |

DB.1

INSERT: 3.4-42-01

fan cooler unit condensate measuring system

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. ~~Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.~~

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE ~~[is] [(or)~~ and ~~air cooler condensate flow rate monitor]~~ [are] instrumented to alarm for increases of 0.5 to 1.0 gpm ~~in the normal flow rates~~. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE. DB.1

Insert: B3.4-36-01

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of $10^{-9}$ μCi/cc radioactivity for particulate monitoring and of $10^{-6}$ μCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate (R-11) and gaseous (R-12) activities because of their sensitivities and rapid responses to RCS LEAKAGE. PA.1

$10^{-11}$

$10^{-7}$

(R-11)

(R-12)

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an

(continued)

**INSERT: B 3.4-86-01**

containment fan cooler unit condensate measuring system

BASES

BACKGROUND (continued)

indicator of potential RCS LEAKAGE. A 1°F increase in dew point is well within the sensitivity range of available instruments.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump ~~(~~and condensate flow from ~~air coolers)~~. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Insert: B3.4-87-01

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. ~~3~~ 2). ~~Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.~~ DB.1

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public.

(continued)

INSERT: B 3.4-87-01

fan cooler unit condensate measuring system

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS leakage detection instrumentation satisfies Criterion 1 of ~~the NRC Policy Statement~~. 10 CFR 50.36

LCO

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

Insert: B.3.4-88-01

Insert: B3.4-88-02

The LCO is satisfied when monitors of diverse flow measurement means are available. Thus, the containment sump monitor, in combination with a gaseous or particulate radioactivity monitor and a containment ~~air cooler condensate flow rate monitor~~, provides an acceptable minimum.

DB.1

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

T.1

ACTIONS

Insert from Page B 3.4-89

A.1 and A.2

With the required containment sump flow monitor inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitor or containment air recirculation unit will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1,

(continued)

INSERT: B 3.4-88-01

fan cooler unit condensate measuring system

INSERT: B 3.4-88-02

The condensate measuring system associated with any one of the fan cooler units satisfies the requirement for a fan cooler unit condensate measuring system.

BASES

ACTIONS

A.1 and A.2 (continued)

must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage.

Restoration of the required sump flow monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

Insert for Page B 3.4-88

The Required Actions ~~A.1 is~~ are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump ~~monitor is~~ and required monitors are inoperable. This allowance is provided because other instrumentation is and detection methods are available to monitor RCS leakage.

T.1

B.1.1, B.1.2, B.2.1, and B.2.2

With both gaseous and particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

Insert: B 3.4-89-01

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere radioactivity monitors. Alternatively, continued operation is allowed if the ~~air cooler condensate flow rate monitoring system~~ is OPERABLE, provided grab samples are taken or water inventory balance performed every 24 hours.

DB.1

The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

~~Required Action B.1 and Required Action B.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous and particulate containment atmosphere radioactivity monitor channel is inoperable. This allowance~~

(continued)

BASES

ACTIONS

~~B.1.1, B.1.2, B.2.1, and B.2.2 (continued)~~

~~is provided because other instrumentation is available to monitor for RCS LEAKAGE.~~

C.1 and C.2

Insert: B3.4-90-01

With the required containment ~~air cooler condensate flow rate monitor~~ inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or a water inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment air ~~cooler condensate flow rate monitor~~ to OPERABLE status.

The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE.

D.1 and D.2

flow

With the required containment atmosphere radioactivity monitor and the required containment ~~air cooler condensate flow rate monitor~~ inoperable, the only means of detecting leakage is the containment sump monitor. This Condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable required monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

E.1 and E.2

If a Required Action of Condition A, B, [C], or [D] cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the

(continued)

INSERT: B 3.4-90-01

fan cooler unit condensate measuring system

BASES

ACTIONS

E.1 and E.2 (continued)

required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.15.3, [SR 3.4.15.4, and SR 3.4.15.5]

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of [18] months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

24

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.

~~2. Regulatory Guide 1.45.~~

2 ~~3~~. FSAR, Section ~~[ 4 ]~~. 6

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.15:
## "RCS Leakage Detection Instrumentation"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.4.15 - RCS Leakage Detection Instrumentation

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-60, Rev.0 (CEOG-12) which makes the LCO 3.0.4 exception applicable to all actions of ITS 3.4.15. An LCO 3.0.4 exception should be applicable to Action D (required containment atmosphere radiation and condensate flow rate monitors inoperable) because other mechanisms (i.e. grab samples, RCS inventory balance, containment sump flow monitor, etc.) exist which are capable of adequately detecting RCS leakage and because a 30 day AOT is usually accompanied by an LCO 3.0.4 exception (e.g., PAM and Remote Shutdown Technical Specifications). As there is already a LCO 3.0.4 exception to Actions A and B and LCO 3.0.4 is not applicable to Action C (which allows indefinite operation), the LCO 3.0.4 exception was moved to apply to all Actions and the specific exceptions for Actions A and B were deleted. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.16:
## "RCS Specific Activity"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS loop average temperature ($T_{avg}$) ≥ 500°F.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm. | ----------- NOTE ------------<br>LCO 3.0.4 is not applicable.<br>------------------------------- | |
| | A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1. | Once per 4 hours |
| | <u>AND</u> | |
| | A.2 Restore DOSE EQUIVALENT I-131 to within limit. | 48 hours |
| B. Gross specific activity of the reactor coolant not within limit of SR 3.4.16.1. | B.1 Be in MODE 3 with $T_{avg}$ < 500°F. | 6 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| C. Required Action and associated Completion Time of Condition A not met.<br><br>OR<br><br>DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1. | C.1 Be in MODE 3 with $T_{avg} < 500°F$. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.16.1 | Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm. | 7 days |
| SR 3.4.16.2 | ------------------NOTE------------------<br>Only required to be performed in MODE 1.<br>----------------------------------------<br><br>Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm. | 14 days<br><br>AND<br><br>Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period |
| SR 3.4.16.3 | ------------------NOTE------------------<br>Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.<br>----------------------------------------<br><br>Determine Ē from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. | 184 days |

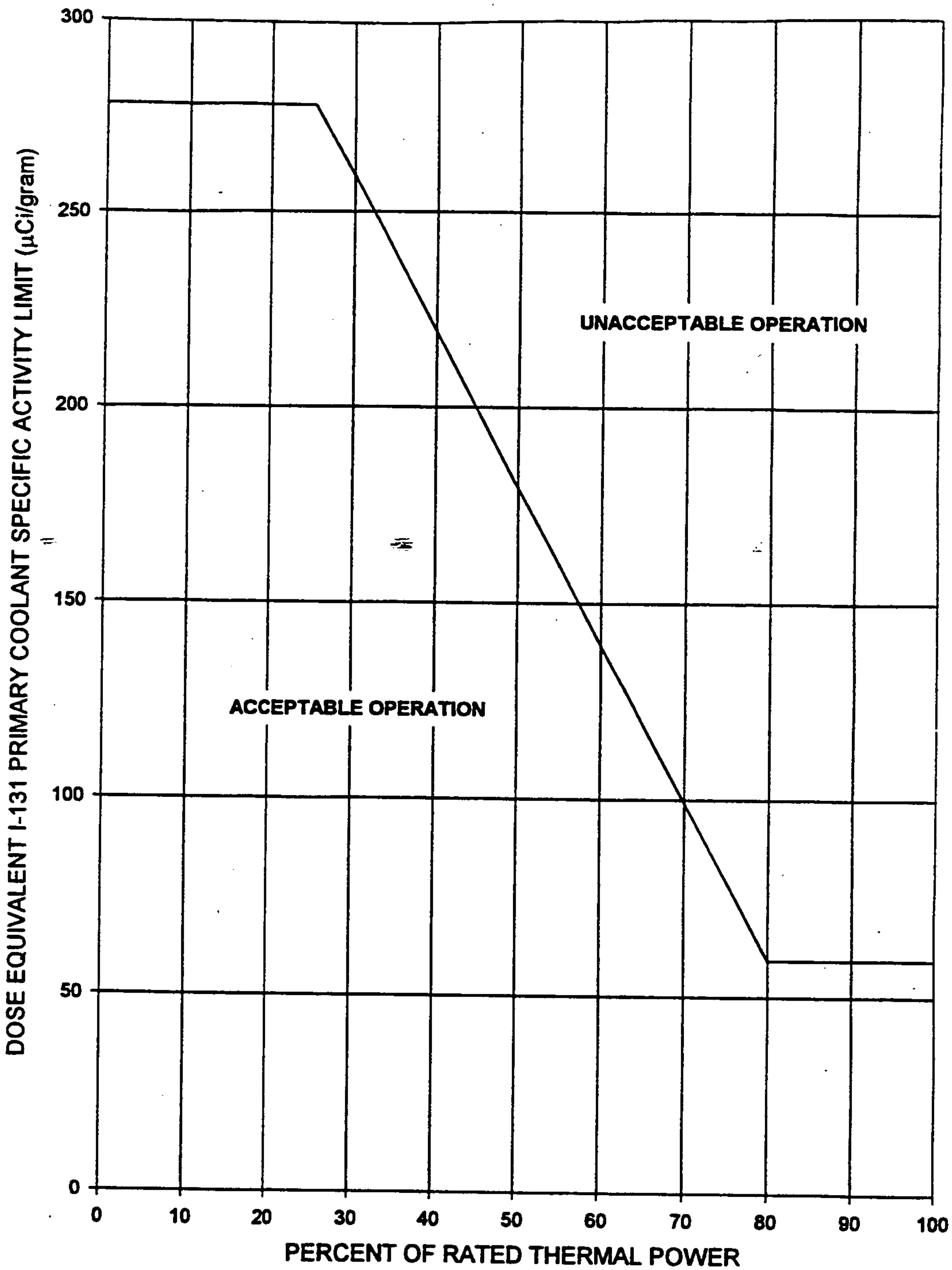


Figure 3.4.16-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-31 Specific Activity
Limit Versus Percent of RATED THERMAL POWER
(Primary Coolant Specific Activity is greater than
1.0 $\mu$Ci/gram DOSE EQUIVALENT I-131)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam

BASES

APPLICABLE SAFETY ANALYSES (continued)

generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.17, "Secondary Specific Activity."

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 50 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of $100/\bar{E}$ μCi/gm for gross specific activity.

The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature $\Delta T$ signal.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG atmospheric dump valves (ADVs) and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the

BASES

APPLICABLE SAFETY ANALYSES (continued)

applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36.

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

BASES

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to establish the trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required to allow operation to continue, if the limit violation resulted from normal iodine spiking.

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1

With the gross specific activity in excess of the allowed

BASES

ACTIONS

B.1 (continued)

limit, the unit must be placed in a MODE in which the requirement does not apply.

Placing the plant in MODE 3 with RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 10 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1 (continued)

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with $T_{avg}$ at least 500°F. The 7 day Frequency considers the low probability of a gross fuel failure during the time.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 10 minutes, excluding iodines and non-gamma emitters. The 10 minute limit on half-lives ensures that Xenon-138 is included in the determination of $\bar{E}$. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.3 (continued)

equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar event.

REFERENCES

1. 10 CFR 100.11, 1973.
2. FSAR, Section 14.2.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.16:
## "RCS Specific Activity"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.1-26 | 121 | 121 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-27 | 121 | 121 | No TSCRs | No TSCRs for this Page | N/A |
| 3.1-28(F 3.1-3) | 121 | 121 | No TSCRs | No TSCRs for this Page | N/A |
| T 4.1-2(1) | 139 | 139 | No TSCRs | No TSCRs for this Page | N/A |
| T 4.1-2(2) | 0 | 0 | No TSCRs | No TSCRs for this Page | N/A |

D. Primary Coolant Activity

LCO 3.4.16 Specification Mode 1 and 2 A.6

Applicability LCO 3.4.16 ~~1.~~ ~~Whenever the reactor is critical~~ or the average reactor coolant temperature is >500°F, the specific activity of the primary coolant shall be limited to:

SR 3.4.16.2 ~~a.~~ ≤1.0 μCi/~~cc~~ Dose Equivalent I-131, gram A.5

and

gross specific activity M2

SR 3.4.1.16.1 ~~b.~~ ≤100/Ē μCi/~~cc~~ ~~for all noble gases with half-lives greater than 10 minutes~~. LA.1

verify every 4 hours SEE CTS T 4.1-2, Notes

Req. Act A.1 Req. Act A.2 ~~2.~~ If the specific activity of the primary coolant is >1.0 μCi/~~cc~~ Dose Equivalent I-131 ~~but~~ within the allowable limit (below and to the left of the line) shown on Figure 3.1-3, operation may continue for up to 48 hours. LCO 3.0.4 not applicable gram A.5 A.3

Condition C Req. Act C.1 ~~3.~~ If the specific activity of the primary coolant is >1.0 μCi/cc Dose Equivalent I-131 for more than 48 hours during one continuous time interval or exceeds the limit line shown on Figure 3.1-3, the reactor shall be ~~immediately~~ brought to the hot shutdown condition with $T_{avg}$ ≤500°F ~~utilizing normal operating procedures~~. in 6 hours M2 A.4

Condition B Req. Act B.1 ~~4.~~ If the specific activity of the primary coolant is >100/Ē μCi/~~cc~~ gm A.5 ~~for all noble gases with half-lives greater than 10 minutes~~, the reactor shall be ~~immediately~~ brought to the hot shutdown condition with $T_{avg}$ ≤500°F ~~utilizing normal operating procedures~~. gross specific activity LA.1 in 6 hours A.4

~~**Bases**~~

~~The limitations on the specific activity of the primary coolant insure that the resulting 2-hour doses at the site boundary will not exceed 1.5 rem to the thyroid and 0.5 rem whole body following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a resultant loss of offsite power. Accident meteorological conditions (5% X/Q) are assumed to exist.~~

~~The action statement permitting Power Operation to continue for limited time periods with the primary coolant's specific activity >1.0 μCi/cc Dose Equivalent I-131, but within the allowable limit shown on Figure 3.1-3, accommodates possible iodine spiking phenomenon which may occur following changes in Thermal Power.~~ A.1

Reducing $T_{avg}$ to <500°F prevents the release of activity, should a steam generator tube rupture, since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Increased surveillance for performing isotopic analyses for iodine is required whenever the Dose Equivalent I-131 exceeds 1.0 μCi/cc and following a significant change in power level to monitor possible iodine spiking phenomenon.

A.1

LCO 3.4.16 – Figure 3.4.16-1

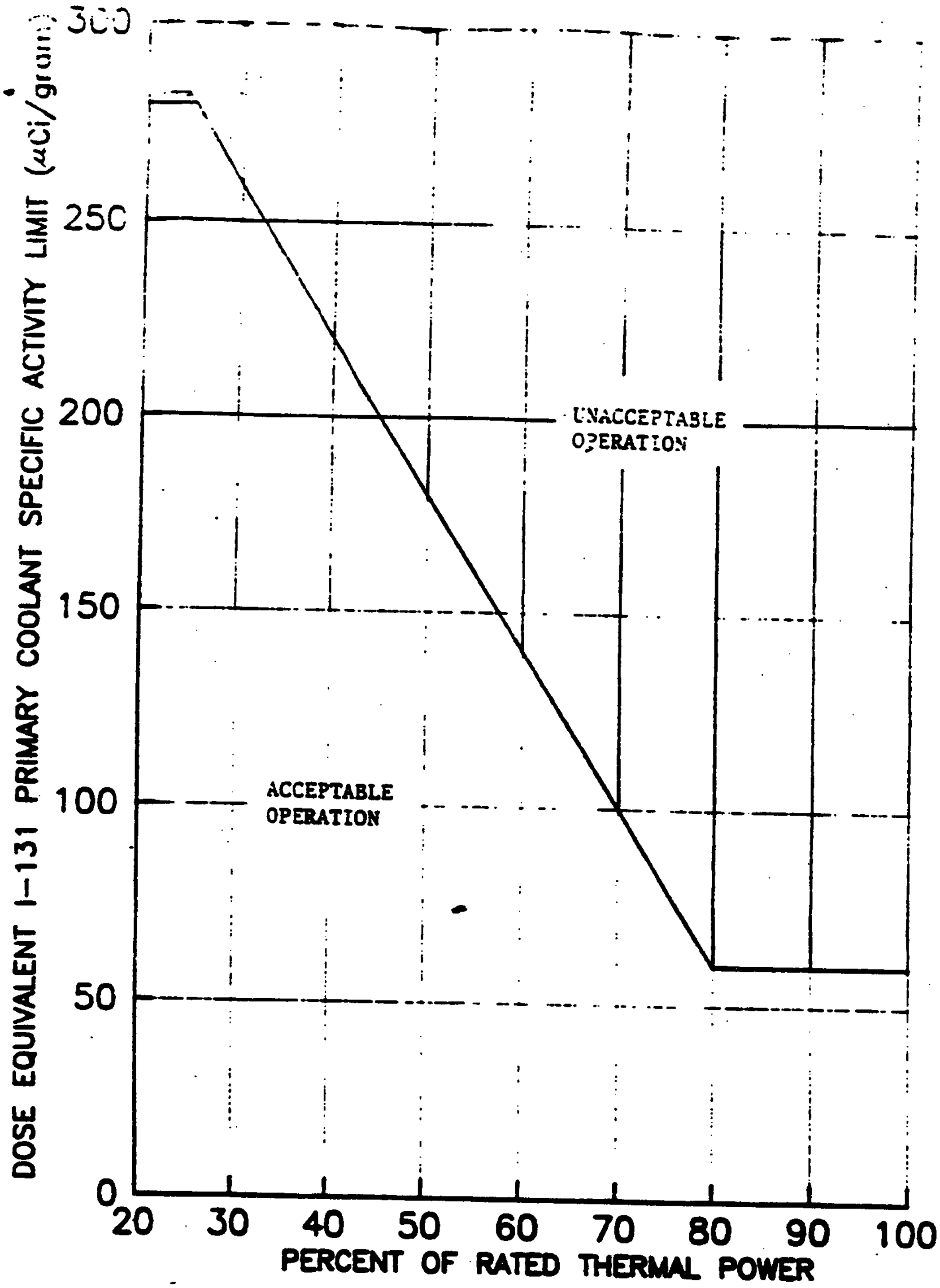


Figure ~~3.1-3~~ 3.4.16-1

**DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0μCi/gram Dose Equivalent I-131**

TABLE 4.1-2 (Sheet 1 of 2)

FREQUENCIES FOR SAMPLING TESTS

| Sample | Analysis | Frequency | Maximum Time Between Analysis |
|---|---|---|---|
| 1. ~~Reactor Coolant~~ SR 3.4.16.1 | ~~Gross Activity~~(3) | ~~5 days/week~~(1)(4) | ~~3 days~~(1) L.3 |
| | Tritium Activity | Weekly(1) | 10 days LA.2 |
| | Boron concentration | 2 days/week | 5 days LA.3 |
| | ~~Radiochemical (gamma)~~(2) ~~Spectral Check~~ | ~~Monthly~~ | ~~45 days~~ L.3 |
| SEE RELOCATED | Oxygen and Chlorides Concentration | 3 times per 7 days | 3 days |
| | Fluorides Concentration | Weekly | 10 days |
| SR 3.4.16.3 | Ē Determination (3) | Semi-Annually 184 days | ~~30 weeks~~ SR 3.0.2 L.1 |
| SR 3.4.16.2 | Isotopic Analysis for I-131, I-133, I-135 | Once per 14 days(3) | ~~20 days~~ SR 3.0.2 M.1 |
| 2. Boric Acid Tank | Boron Concentration, Chlorides | Weekly | 10 days |
| 3. Spray Additive Tank | NaOH Concentration | Monthly | 45 days |
| 4. Accumulators | Boron Concentration | Monthly | 45 days |
| 5. Refueling Water Storage Tank | Boron Concentration pH, Chlorides | Monthly | 45 days |
| | Gross Activity | Quarterly | 16 weeks |
| 6. Secondary Coolant | I-131 Equivalent (Isotopic Analysis) | Monthly | 45 days |
| | Gross Activity | 3 times per 7 days | 3 days |
| 7. Component Cooling Water | Gross Activity, Corrosion Inhibitor and pH | Monthly | 45 days |
| 8. Spent Fuel Pool (when fuel stored) | Gross Activity Boron Concentration, Chlorides | Monthly | 45 days |

SEE CTS MASTER MARKUP

TABLE 4.1-2 (Sheet 2 of 2)

FREQUENCIES FOR SAMPLING TESTS

FOOTNOTES:

SR 3.4.16.1

(1) ~~A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of $\mu$Ci/cc.~~

(2) ~~A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 10 minutes making up at least 95% of the total activity of the primary coolant.~~

LA.1

M2

L.3

SR 3.4.16.3

~~(3)~~ $\bar{E}$ determination will be ~~started when the gross activity analysis indicates $\geq$ 10 $\mu$Ci/cc and will be redetermined if the primary coolant gross radioactivity changes by more than 10 $\mu$Ci/cc in accordance with Specification 3.1.D~~

L.2

L.4

SEE RELOCATED

(4) Whenever the Gross Failed Fuel Monitor is inoperable, the sampling frequency shall be increased to twice per day, five days per week. The maximum time between analyses shall be sixteen hours for the two samples taken on a given day and three days between daily analysis. This accelerated sampling frequency need only be performed until the Gross Failed Fuel Monitor is declared operable.

Req. Act A.1

SR 3.4.16.2 Frequency

(5) Once per 4 hours whenever the DOSE EQUIVALENT I-131 exceeds 1.0 $\mu$Ci/cc or one sample after two hours but before six hours following a thermal power change exceeding 15 percent of the rated thermal power within a one-hour period.

Note to SR 3.4.16.3

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.16:
## "RCS Specific Activity"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.4.16 - RCS Specific Activity

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases that are designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 ITS 3.4.16, Required Action A.1, includes a Note that LCO 3.0.4 is not applicable to specific activity limits for Dose Equivalent I-131. This note will allow entering the Applicable Modes for LCO 3.4.16 during a startup even if specific activity limits for Dose Equivalent I-131 are not met. This change is needed because it recognizes that transient specific activity excursions can occur during normal plant startups and that the plant has the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

This change is acceptable because of the significant conservatism incorporated into the specific activity limit, the low probability of an event for which specific activity is a limiting factor in the consequences, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation. This is an administrative change with no impact on safety because there is no equivalent to LCO 3.0.4 in the CTS; therefore, providing an exception results in no changes to the existing requirements. The justification for adding ITS LCO 3.0.4 is addressed on Discussion of Changes for ITS Section 1.0.

A.4 CTS 3.1.D.3 and CTS 3.1.D.4 specifies that the reactor shall be "immediately" brought to the hot shutdown condition with Tavg ≤500°F, "utilizing normal operating procedures" for failure to meet specific activity limits. Under the same conditions, ITS 3.4.16, Required Action B.1 and ITS 3.4.16, Required Action C.1, specify the completion time to be in Mode 3 with Tavg < 500°F as "within 6 hours." The change to the completion time is an administrative change with no impact on safety because 6 hours is reasonable, based on operating experience, to reach Mode 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

A.5 CTS 3.1.D.1 establishes primary coolant activity levels in units of μCi per cubic centimeter for both Dose Equivalent I-131 and gross specific activity. ITS LCO 3.4.16 maintains the same numerical limits but expresses those limits in units of μCi per gram. This change is needed and is acceptable because Dose Equivalent I-131 and gross specific activity are determined at ambient temperatures and pressures where one gram of water is equivalent to one cubic centimeter of water. Additionally, offsite dose calculations are based on limits expressed in units of μCi per gram. Therefore, this is an administrative change with no impact on safety.

A.6 CTS 3.1.D.1 specifies that limits for primary coolant activity are applicable whenever the reactor is critical or the average reactor coolant temperature is > 500 °F. ITS LCO 3.4.16 specifies that limits for primary coolant activity are applicable in Modes 1 and 2 and in

Mode 3 with RCS average temperature (Tavg) ≥ 500°F. However, minimum temperature for criticality limits which are significantly greater than 500 °F ensure that CTS requirements for primary coolant activity are applicable before the CTS allows the reactor to be placed in a status equivalent to the ITS Mode 2. Therefore, this is an administrative change with no impact on safety.

MORE RESTRICTIVE

M.1 CTS Table 4.1-2, Item 1, establishes a surveillance Frequency for Isotopic Analysis for I-131, I-133, I-135 as once per 14 days with a "maximum time between analysis" of 20 days. ITS SR 3.4.16.2 also requires verification of reactor coolant dose equivalent I-131 specific activity every 14 days but the limit for the maximum time between analyses is based on ITS SR 3.0.2 which allows a 25% grace period (i.e., the maximum interval is 17.5 days). This change is not needed to satisfy technical requirements but is being adopted for consistency with the NUREG-1431 and to simplify application of ITS SR 3.0.2. This change has no impact on safety.

M.2 CTS 3.1.D.1 specifies that the acceptance criteria for reactor coolant gross activity, a function of $\bar{E}$, is limited to "noble gases with half-lives greater than 10 minutes." This is consistent with CTS 1.14, the definition of $\bar{E}$-Average Disintegration Energy, which limits $\bar{E}$ to the Noble gas $\bar{E}$.

ITS LCO 3.4.16 and the acceptance criteria for ITS 3.4.16.1 are based on the ITS Definition, $\bar{E}$-Average Disintegration Energy. ITS LCO 3.4.16 and the acceptance criteria for ITS 3.4.16.1 are based on the ITS Definition, $\bar{E}$-Average Disintegration Energy. The ITS Definition of $\bar{E}$ differs from the CTS definition in that the ITS definition includes all isotopes (not just Noble gases) in the reactor coolant, other than iodines, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant. This change, including all isotopes except iodines when measuring gross specific activity, is needed because the ITS definition ensures that contributions from isotopes other than Noble gases, although typically not significant, are counted. (Maintaining the CTS allowance permitting the exclusion of isotopes with half lives > 10 minutes rather than adopting the ITS allowance permitting the exclusion of isotopes with half lives > 15 minutes is needed to ensure that Xenon-138 is included in $\bar{E}$-Average Disintegration Energy consistent with current analysis assumptions.) This change, excluding iodines from the definition of $\bar{E}$ and gross specific activity, is acceptable because the dose contribution of iodines are limited by the ITS SR 3.4.16.2 limits for Dose Equivalent I-131. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS Table 4.1-2, Item 1, establishes a surveillance Frequency for E Bar determination as semi-annually with a "maximum time between analysis" of 30 weeks. ITS SR 3.4.16.3 also requires verification of E Bar every 184 days but the maximum time between analyses is based on ITS SR 3.0.2 which allows a 25% grace period for a maximum interval of approximately 32.5 weeks. This change is not needed to satisfy technical requirements but is being adopted for consistency with the NUREG-1431 and to simplify application of ITS SR 3.0.2. Extensive experience has shown that E Bar does not change rapidly. Additionally, unexpected changes in E Bar would be evident from changes in other primary coolant activity levels which are monitored more frequently. Therefore, keeping the normal Frequency for E Bar determination as 184 days but extending the maximum time between analyses from 30 weeks to 32.5 weeks has no significant adverse impact on safety.

L.2 CTS Table 4.1-2, Item 1, establishes a surveillance Frequency for E Bar determination as semi-annually. This SR Frequency is modified by CTS Table 4.1-2, Note 3, which specifies that E Bar determination will be started when the gross activity analysis indicates $\geq$ 10 $\mu$Ci/cc. This modification of the SR Frequency is intended to allow determination of E Bar to be deferred until plant conditions are such that meaningful results can be obtained.

ITS SR 3.4.16.3 also requires verification of E Bar every 184 days but provides a more precise method of ensuring that the sample is taken only when plant conditions are established so that the sample provides an accurate indication of plant conditions. SR 3.4.16.3 ensures that appropriate plant conditions are established by requiring that the E Bar verification can be made only in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. A Note to SR 3.4.16.3 allows deferring performance of the SR until these conditions can be established.

The combination of the sampling restriction in the SR and the allowance provided in the SR note ensure that the sample is accurate by both allowing and requiring the SR be performed when radioactive materials are at equilibrium so the analysis results are representative of actual plant conditions and can be trended. This change has no impact on safety because the combination of the allowance for deferral of the SR (the SR Note) and restrictions about the conditions for sampling ensure that the SR provides an accurate indication of actual plant conditions.

L.3 CTS Table 4.1-2, Item 1, requires verification at least five days per week of the of "gross activity" and requires verification every month of gross specific activity using a "Radiochemical (gamma) Spectral Check." Gross activity and Radiochemical (gamma) Spectral Check are defined in Footnotes 1 and 2 of CTS Table 4.1-2. The Radiochemical (gamma) Spectral Check is equivalent (See ITS 3.4.16, DOC M.2) to the gross specific activity defined in the Based of ITS 3.4.16.

ITS SR 3.4.16.1 requires verification every 7 days of the gross specific activity. This change requires more Frequent verification (every 7 days versus monthly) of the gross specific activity (See ITS 3.4.16. DOC M.2) and eliminates the explicit requirement to verify gross activity at least five days per week.

This change is acceptable because this check was intended to provide an indication of fuel failure by monitoring for an increase in gross activity. Extending the SR Frequency from five days per week to once per week is acceptable because industry experience demonstrates there is a low probability of significant fuel failure that is not readily apparent by other indications, industry experience indicates that trending of results for gross activity determinations at a 7 day Frequency is effective in identifying incipient fuel failure prior to exceeding limits, and the operation of the gross failed fuel monitor which is required by plant licensee programs controlled outside of Technical Specifications (See Relocated Item R.21). This change has no significant adverse impact on safety because the combination of the low probability of fuel failure, the use of trending to identify incipient fuel failure prior to exceeding limits, and the presence of a gross failed fuel monitor provide a high degree of assurance that failed fuel will be detected in a timely manner.

L.4 CTS Table 4.1-2, Note 3, specifies that E Bar will be redetermined if the primary coolant gross radioactivity changes by more than 10 $\mu$Ci/cc. ITS SR 3.4.16.3 does not include this requirement.

This change is needed to ensure that E Bar measurements are not skewed by a crud burst or other similar event. This change is acceptable because extensive industry experiences indicates that E Bar changes slowly and the combination of the sampling restrictions in the SR and the allowance provided in the SR note ensure that the sample is accurate by both allowing and requiring the SR be performed when radioactive materials are at equilibrium so the analysis results are representative of actual plant conditions and can be trended. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

LA.1 CTS 3.1.D specifies that limits on specific activity apply only to "noble gases with half-lives greater than 10 minutes." ITS LCO 3.4.16 establishes limits for gross specific activity (See ITS 3.4.16, DOC M.2) with the clarification in the ITS Bases that gross specific activity is basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken.

These descriptions of what constitutes a gross specific activity determination are not retained in ITS LCO 3.4.16 and are moved to the Bases. This change is acceptable because ITS LCO 3.4.16 maintains the requirement that reactor coolant activity levels be maintained within the specified limits. Maintaining this information in the Bases is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, and ITS 5.5.13, Technical Specifications (TS) Bases Control Program, are designed to assure that changes to the ITS Bases do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

LA.2 CTS Table 4.1-2, Item 1, includes a surveillance for a weekly measurement of tritium activity in the reactor coolant. ITS 3.4.16 does not retain these requirements which are being relocated to the Offsite Dose Calculation Manual (ODCM).

This change is acceptable because the neither the CTS nor the ITS include any Limiting Conditions for Operation or acceptance criteria associated with tritium activity in the reactor coolant or the radiochemical spectrum of the coolant. This tritium activity in the reactor coolant and the radiochemical spectrum of the coolant are associated with limits that are currently maintained in the ODCM. Therefore, relocating these requirements to the ODCM does not eliminate or reduce any requirements either in the Technical Specifications or the ODCM.

Maintaining these requirements in the ODCM is acceptable because the ODCM is approved by the NRC prior to implementation and any change to the ODCM is controlled in accordance with ITS 5.5.1. ITS 5.5.1.a provides for regulatory oversight of changes to the ODCM by requiring that a determination that the change(s): a) maintains the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I; and, b) does not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations. Additionally, ITS 5.5.1.c requires that all changes to the ODCM be submitted to the NRC with the Radioactive Effluent Release Report required by ITS 5.6. Therefore, moving requirements for tritium and radiochemical spectrum to the ODCM does not change any existing requirement and ITS 5.5.1 provides an appropriate change control process for the ODCM. Therefore, this change has no significant adverse impact on safety.

LA.3 CTS Table 4.1-2, Item 1, includes a surveillance for a twice weekly measurement of boron concentration. ITS 3.4.16 does not retain this requirement which is being relocated to plant procedures.

Maintaining requirements for measurement of boron concentration outside of Technical Specifications is acceptable because boron concentration is an intrinsic part of the verification that shutdown margin and control rod insertion limits are met. ITS Section 3.1, Reactivity Control Systems, and 3.9, Refueling Operations, maintain requirements for the verification of shutdown margin and rod insertion limits and these requirements ensure that boron concentration is adequately monitored. Therefore, this change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.16:
## "RCS Specific Activity"

## PART 4:

## No Significant Hazards Considerations
## for
## Changes between CTS and ITS
## that are
## Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.4.16 - RCS Specific Activity

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change extends the maximum grace period for the performance of semi-annual determination of E Bar from 30 weeks to 32.5 weeks to be consistent with the 25% grace period allowed by ITS SR 3.0.2. This change will not result in a significant increase in the probability of an accident previously evaluated because the Frequency for the determination of E Bar is not related to the precursor of any analyzed accident. This change will not result in a significant increase in the consequences of an accident previously evaluated because extensive experience has shown that E Bar does not change rapidly. Additionally, unexpected changes in E Bar would be evident from changes in other primary coolant activity levels which are monitored more frequently. Therefore, keeping the normal Frequency for E Bar determination as 184 days but extending the maximum time between analyses from 30 weeks to 32.5 weeks has no significant adverse impact on safety.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change to the method for determining E Bar. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because extensive experience has shown that E Bar does not change rapidly. Additionally, unexpected changes in E Bar would be evident from changes in other primary coolant activity levels which are monitored more frequently. Therefore, keeping the normal Frequency for E Bar determination as 184 days but extending the maximum time between analyses from 30 weeks to 32.5 weeks has no significant adverse impact on safety.

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LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change modifies the plant conditions that require initiation of periodic verification of E Bar from whenever gross activity analysis indicates $\geq$ 10 $\mu$Ci/cc to the ITS plant conditions as follows: 31 days after a minimum of 2 effective full power days and 20 days of Mode 1 operation have elapsed since the reactor was last subcritical for $\geq$ 48 hours

   This change will not result in a significant increase in the probability of an accident previously evaluated because the Frequency for the determination of E Bar is not related to the precursor of any analyzed accident. This change will not result in a significant increase in the consequences of an accident previously evaluated because the ITS still requires verification of E Bar every 184 days but provides a more precise method of ensuring that the sample is taken only when plant conditions are established so that the sample provides an accurate indication of plant conditions. The combination of the sampling restriction in the SR and the allowance provided in the SR note ensure that the sample is accurate by both allowing and requiring the SR be performed when radioactive materials are at equilibrium so the analysis results are representative of actual plant conditions and can be trended. This change has no impact on safety because the combination of the allowance for deferral of the SR (the SR Note) and restrictions about the conditions for sampling ensure that the SR provides an accurate indication of actual plant conditions.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal plant operation are consistent with the current safety analysis assumptions because there is no change in the method used to determine E Bar. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the ITS still requires verification of E Bar every 184 days but provides a more precise method of ensuring that the sample is taken only when plant conditions are established so that the sample provides an accurate indication of plant conditions. The combination of the sampling restriction in the SR and the allowance provided in the SR note ensure that the sample is accurate by both allowing and requiring the SR be performed when radioactive materials are at equilibrium so the analysis results are representative of actual plant conditions and can be trended. This change has no impact on safety because the combination of the allowance for deferral of the SR (the SR Note) and restrictions about the conditions for sampling ensure that the SR provides an accurate indication of actual plant conditions.

---

LESS RESTRICTIVE
<u>("L.3" Labeled Comments/Discussions)</u>

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change requires more Frequent verification (every 7 days versus monthly) of the gross specific activity (See ITS 3.4.16. DOC M.2) and eliminates the explicit requirement to verify gross activity at least five days per week. This change will not result in a significant increase in the probability of an accident previously evaluated because the Frequency for the determination of gross activity is not related to the precursor of any analyzed accident. This change will not result in a significant increase in the consequences of an accident previously evaluated because this check was intended to provide an indication of fuel failure by monitoring for an increase in gross activity. Extending the SR Frequency from five days per week to once per week is acceptable because industry experience demonstrates there is a low probability of significant fuel failure that is not readily apparent by other indications, industry experience indicates that trending of results for gross activity determinations at a 7 day Frequency is effective in identifying incipient fuel failure prior to exceeding limits, and the operation of the gross failed fuel monitor which is required by plant licensee programs controlled outside of Technical Specifications (See Relocated Item R.21). This change has no significant adverse impact on safety because the combination of the low probability of fuel failure, the use of trending to identify incipient fuel failure prior to exceeding limits, and the presence of a gross failed fuel monitor provide a high degree of assurance that failed fuel will be detected in a timely manner.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because the method used to determine gross activity will not change.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because this check was intended to provide an indication of fuel failure by monitoring for an increase in gross activity. Extending the SR Frequency from five days per week to once per week is acceptable because of the following: extensive industry experience there is a low probability of significant fuel failure that is not readily apparent by other indications, industry experience indicates that trending of results for gross activity determinations at a 7 day Frequency is effective in identifying incipient fuel failure prior to exceeding limits, and the operation of the gross failed fuel monitor which is required by plant licensee programs controlled outside of Technical Specifications (See Relocated Item R.21). This change has no significant adverse impact on safety because the combination of the low probability of fuel failure, the use of trending to identify incipient fuel failure prior to exceeding limits, and the presence of a gross failed fuel monitor provide a high degree of assurance that failed fuel will be detected in a timely manner.

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LESS RESTRICTIVE
("L.4" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change eliminates the requirement that E Bar be redetermined if the primary coolant gross radioactivity changes by more than 10 $\mu$Ci/cc. This change will not result in a significant increase in the probability of an accident previously evaluated because the Frequency for the determination of E Bar is not related to the precursor of any analyzed accident. This change will not result in a significant increase in the consequences of an accident previously evaluated because the change will ensure that E Bar measurements are not skewed by a crud burst or other similar event. This change is acceptable because extensive industry experiences indicates that E Bar changes slowly and the combination of the sampling restrictions in the SR and the allowance provided in the SR note ensure that the sample is accurate by both allowing and requiring the SR be performed when radioactive materials are at equilibrium so the analysis results are representative of actual plant conditions and can be trended.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change to the method for determining E Bar. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the change will ensure that E Bar measurements are not skewed by a crud burst or other similar event. This change is acceptable because extensive industry experiences indicates that E Bar changes slowly and the combination of the sampling restrictions in the SR and the allowance provided in the SR note ensure that the sample is accurate by both allowing and requiring the SR be performed when radioactive materials are at equilibrium so the analysis results are representative of actual plant conditions and can be trended.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.4.16:
## "RCS Specific Activity"

## PART 5:

## NUREG-1431
## Annotated to show differences between
## NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.4.16

This ITS Specification is based on NUREG-1431 Specification No. **3.4.16**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| CEOG-045 | **137 R0** | RELOCATION OF THE 3.4.16 ACTION NOTE A TO BASES | Approved by NRC | Incorporated | **T.2** |
| WOG-001.2 | **003 R1** | RELOCATE REFERENCES TO THYROID DOSE CONVERSION FACTORS TO THE BASES. | Rejected by NRC | Not Incorporated. | **N/A** |
| WOG-015 | **028 R0** | DELETE UNNECESSARY ACTION TO MEASURE GROSS SPECIFIC ACTIVITY | Approved by NRC | Incorporated | **T.1** |

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.16 RCS Specific Activity

<3.1.D.1>
<DOC M2>

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

<3.1.D.1>
<DOC A.6>

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average loop temperature ($T_{avg}$) ≥ 500°F.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| <DOC A.3> <3.1.D.1.a> A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm. | ------------Note------------ LCO 3.0.4 is not applicable. ---------------------------- | |
| <3.1.D.2> <Table 4.1-2, Note 5> | A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1. | ~~Once per 4 hours~~ |
| | AND | |
| <3.1.D.2> | A.2 Restore DOSE EQUIVALENT I-131 to within limit. | ~~48 hours~~ |
| <3.1.D.4> <DOC A.4> B. Gross specific activity of the reactor coolant not within limit of SR 3.4.16.1. | ~~B.1 Perform SR 3.4.16.2.~~ | ~~4 hours~~ T.1 |
| | ~~AND~~ | |
| | B.~~2~~ 1 Be in MODE 3 with $T_{avg}$ < 500°F. | 6 hours |

(continued)

<CTS>

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| <3.1.D.3> <DOC A.4> C. Required Action and associated Completion Time of Condition A not met.<br><br>OR<br><br><3.1.D.3> <DOC A.4> DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1. | C.1 Be in MODE 3 with $T_{avg} < 500°F$. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| <3.1.D.1.b> <T 4.1-2, #1> <DOC M.2> <DOC L.3> SR 3.4.16.1 Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm. | 7 days |
| <3.1.D.1.a> <Table 4.1-2, Item 1> SR 3.4.16.2 ------------------NOTE------------------ Only required to be performed in MODE 1. ---------------------------------------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm. | 14 days<br><br>AND<br><br><Table 4.1-2 Item 1, Note 5> Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.4.16.3 ------------------NOTE------------------<br>Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.<br>-----------------------------------------------------<br>Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. | <br><br><br>184 days |

<DOC L.2>

<Table 4.1-2, Item 1>
<DOC L.1>
<DOC L.2>
<DOC L.4>

DB.1

Insert: 3.4-46-01

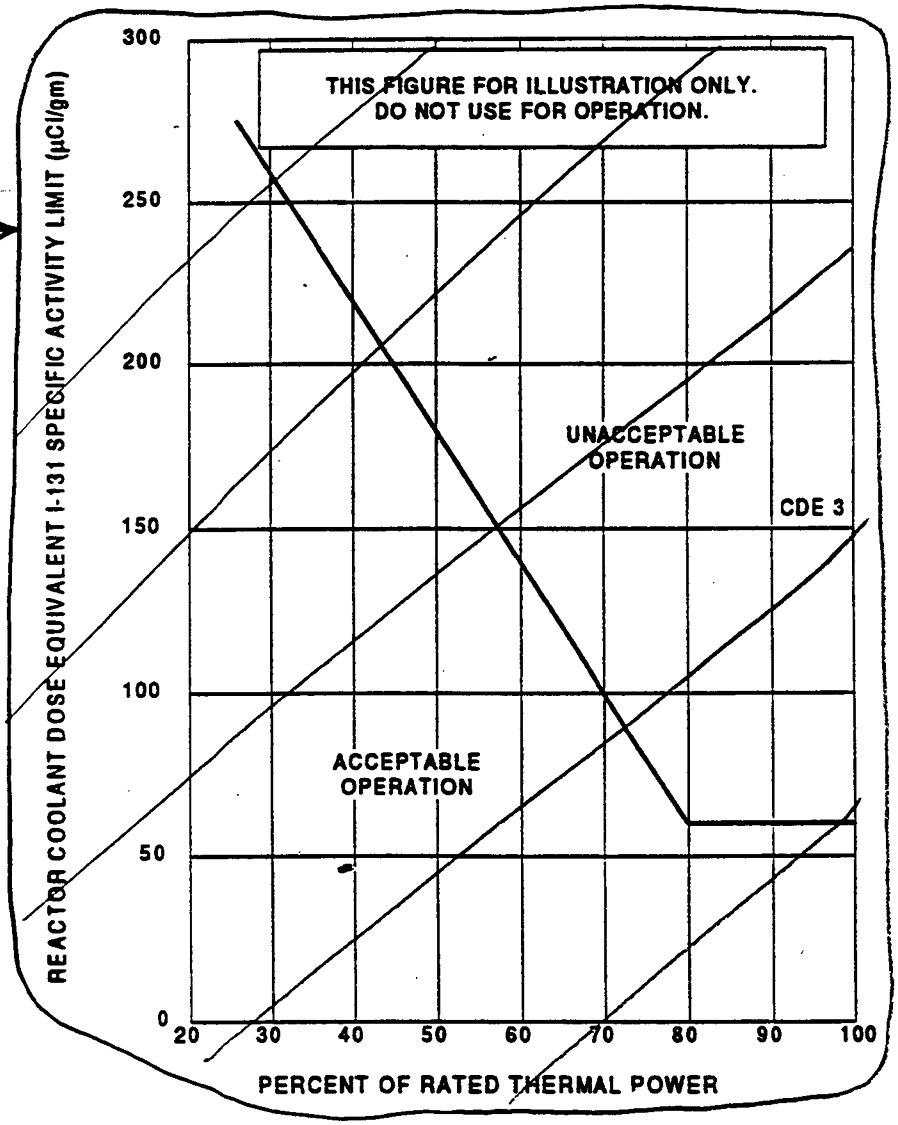


Figure 3.4.16-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity
Limit Versus Percent of RATED THERMAL POWER

INSERT: B 3.4-46-01

Insert a clean copy of CTS Figure 3.1-1 from CTS Page 3.1-28.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.~~6~~ 17, "Secondary Specific Activity." PA.1

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 50 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/Ē μCi/gm for gross specific activity.

The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the ~~SG power operated relief valves~~ and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

Atmospheric Dump Valves (ADVs)

PA.1

DA1

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR

(continued)

BASES

| | |
|---|---|
| APPLICABLE SAFETY ANALYSES (continued) | accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.<br><br>The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.<br><br>RCS specific activity satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36. PA.1 |
| LCO | The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.<br><br>The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits. |
| APPLICABILITY | In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.<br><br>For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves. |

(continued)

BASES (continued)

ACTIONS

Required Actions of Condition A

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

T.2

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to ~~continue to provide a~~ trend.

establish the

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

to allow operation to continue

B.1 ~~and B.2~~

With the gross specific activity in excess of the allowed limit, ~~an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample.~~

T.1

Insert: B 3.4-96-01

~~The change within 6 hours to MODE 3 and~~ RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

Placing the plant in MODE 3 with

(continued)

**INSERT: B 3.4-96-01**

the unit must be placed in a MODE in which the requirement does not apply.

BASES

ACTIONS (continued)

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity. 10 CLB.1

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with $T_{avg}$ at least 500°F. The 7 day Frequency considers the ~~unlikelihood~~ of a gross fuel failure during the time. low probability PA.1

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.3

A radiochemical analysis for E determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The E determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for E is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes E does not change rapidly.

and non-gamma emitters

10

CLB.1

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for E is representative and not skewed by a crud burst or other similar ~~abnormal~~ event.

DB.1

REFERENCES

1. 10 CFR 100.11, 1973.
2. FSAR, Section ~~15.6.3~~. 14.2

The 10 minute limit on half-lives ensures that Xenon-138 is included in the determination of E.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.4.16:
# "RCS Specific Activity"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.4.16 - RCS Specific Activity

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 The ITS Definition of $\bar{E}$ differs from the CTS definition in that the ITS definition includes all isotopes (not just Noble gases) in the reactor coolant, other than iodines, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant. This change, including all isotopes except iodines when measuring gross specific activity, is needed because the ITS definition ensures that contributions from isotopes other than Noble gases, although typically not significant, are counted. (Maintaining the CTS allowance permitting the exclusion of isotopes with half lives > 10 minutes rather than adopting the ITS allowance permitting the exclusion of isotopes with half lives > 15 minutes is needed to ensure that Xenon-138 is included in $\bar{E}$-Average Disintegration Energy consistent with current analysis assumptions.) This change, excluding iodines from the definition of $\bar{E}$ and gross specific activity, is acceptable because the dose contribution of iodines are limited by the ITS SR 3.4.16.2 limits for Dose Equivalent I-131. Therefore, this change has no significant adverse impact on safety.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-028 (WOG-15), which deletes Required Action B.1, which requires performance of SR 3.4.16.2, measurement of dose equivalent I-131 every 4 hours while the plant is being shutdown because gross activity is not within limits. This change is acceptable because this action is an unnecessary burden as the plant is required to be in Mode 3 with Tavg < 500 F within 6 hours at which time ITS LCO 3.4.16 is no longer applicable. SR 3.4.16.2 must be performed in order to verify restoration of the specific activity to within limits and is not otherwise required while the plant is being shutdown. Additionally, if the Condition B is entered and the plant is in MODE 2 in 4 hours or less, the NUREG Required Action is in conflict with the NOTE of SR 3.4.16.2 which states that this SR is only required in Mode 1. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

T.2 This change incorporates Generic Change TSTF-137 (CEOG-45), which revises the Bases to move the description of the Note to Required Action A.1. This is an administrative change. This generic change to NUREG 1431, Rev. 1, has been approved by the NRC.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point
3

Docket # 50-286
Accession # 9812150197
Date 12/11/98 of Ltr
Regulatory Docket File

# Improved

# Technical Specifications

# Conversion Submittal

Volume 9

New York Power
Authority

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.5.1:
# "Accumulators"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.1 Accumulators

LCO 3.5.1 Four ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with reactor coolant system pressure > 1000 psig.

-----NOTES-----

1. In MODE 3, all accumulator discharge isolation valves may be closed and energized for up to 8 hours during the performance of reactor coolant system hydrostatic testing.

2. In MODE 3, one accumulator discharge isolation valve may be closed and energized for up to 8 hours for accumulator check valve leakage testing.

---

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One accumulator inoperable due to boron concentration not within limits of SR 3.5.1.4. | A.1 Restore boron concentration to within limits of SR 3.5.1.4. | 72 hours |
| B. One accumulator inoperable for reasons other than Condition A. | B.1 Restore accumulator to OPERABLE status. | 1 hour |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| C. Required Action and associated Completion Time of Condition A or B not met. | C.1 Be in MODE 3.<br><br>AND<br><br>C.2 Reduce reactor coolant system pressure to ≤ 1000 psig. | 6 hours<br><br><br><br>12 hours |
| D. Two or more accumulators inoperable. | D.1 Enter LCO 3.0.3. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.1.1 | Verify each accumulator discharge isolation valve is fully open. | 12 hours |
| SR 3.5.1.2 | Verify borated water volume in each accumulator is ≥ 775 cubic feet and ≤ 815 cubic feet. | 12 hours |
| SR 3.5.1.3 | Verify nitrogen cover pressure in each accumulator is ≥ 600 psig and ≤ 700 psig. | 12 hours |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.1.4 | Verify boron concentration in each accumulator is $\geq$ 2000 ppm and $\leq$ 2600 ppm. | 31 days<br><br>AND<br><br>-----NOTE------<br>Only required to be performed for affected accumulators<br>--------------<br><br>Once within 6 hours after each solution volume increase of $\geq$ 3 cubic feet, 10 % of indicated level, that is not the result of addition from the refueling water storage tank |
| SR 3.5.1.5 | Verify power is removed from each accumulator isolation valve operator when reactor coolant system pressure is $\geq$ 2000 psig. | 31 days |

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for any LOCA that reduces RCS pressure to below the accumulator pressure.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series.

BASES

BACKGROUND (continued)

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During

BASES

APPLICABLE SAFETY ANALYSES (continued)

this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and high head safety injection (HHSI) pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the HHSI pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the

BASES

APPLICABLE SAFETY ANALYSES (continued)

accumulators, since the accumulators are emptied, once discharged.

Accumulator tank size and accumulator water volume directly affect the volume of nitrogen cover gas whose expansion produces the passive injection and thus affects injection rate. The amount of water is also important since the accumulator water which has not been injected and bypassed during blowdown is primarily responsible for filling the lower plenum (refill) and downcomer. The elevation head of the downcomer water provides the driving force for core reflooding (Ref. 3).

For large break LOCAs, changes in accumulator water volume can result in either improved or worsened analysis results; therefore, a nominal accumulator water volume of 795 cubic feet is modeled in the analysis (Ref. 3).

For small break LOCAs, changes in accumulator water volume are not significant because the clad temperature transient is terminated before the accumulators empty; therefore, a nominal accumulator water volume of 795 cubic feet is modeled in the analysis (Ref. 3).

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen

BASES

APPLICABLE SAFETY ANALYSES (continued)

cover pressure limit prevents injection of nitrogen into the RCS, accumulator relief valve actuation, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 3 and 4).

The accumulators satisfy Criterion 3 of 10 CFR 50.36.

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 2000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

BASES

APPLICABILITY (continued)

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated discharge isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

Note 1 provides an exception to SR 3.5.1.1 and SR 3.5.1.5 and specifies that all accumulator discharge isolation valves may be closed and energized for up to 8 hours during the performance of reactor coolant system hydrostatic testing. This allowance is necessary because limits imposed by the Pressure/Temperature Limits for a hydrostatic leak test, could, in some instances, require reactor coolant system hydrostatic testing above 350°F (Mode 3). This allowance is acceptable because hydrostatic testing is performed in MODE 3 when the need for the accumulators is reduced and Note 1 limits the duration to the time needed to perform required testing.

Note 2 also provides an exception to SR 3.5.1.1 and SR 3.5.1.5 and specifies that one accumulator discharge isolation valve may be closed and energized in MODE 3 for up to 8 hours for accumulator check valve leakage testing. This allowance is acceptable because testing is limited to MODE 3 when the need for the accumulators is reduced and Note 2 limits the duration to the time needed to perform required testing.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available

BASES

ACTIONS

A.1 (continued)

ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and reactor coolant pressure reduced to ≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator valve should be verified to be fully open every 12 hours. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If a discharge isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling the affected accumulator within 6 hours after an increase of 8.4 cubic feet will identify whether inleakage has caused a reduction in boron concentration to below the required limit. Considering the nominal accumulator volume of 795 cubic feet of water, inleakage of 8.4 cubic feet of pure water would result in a boron concentration reduction of approximately 1%. An increase in the accumulator volume of 8.4 cubic feet causes a change of approximately 10% in the indicated accumulator level. It is not necessary to verify boron concentration if the added water inventory is from the refueling

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.4 (continued)

water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 4).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator discharge isolation valve operator when the pressurizer pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated discharge isolation valves when pressurizer pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

Should closure of a valve occur, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. FSAR, Chapter 6.

2. 10 CFR 50.46.

3. FSAR, Chapter 14.

4. NUREG-1366, February 1990.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.1:
## "Accumulators"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.3-2 | 154 | 154 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-3 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-4 | 139 | 139 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-5 | 53 | 53 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-14 | 132 | 132 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-15 | 139 TSCR 97-175 | 139 TSCR 97-175 | IPN 97-175 | Changes to Bases Pages | |
| 3.3-16 | 154 | | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-17 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| T 4.1-1(3) | 168 TSCR 98-043 | 168 TSCR 98-043 | IPN 98-043 | Instrument Channel Surveillance Intervals Extended to 24 Months | Incorporated |
| T 4.1-1(6) | 181 TSCR 98-043 | 181 TSCR 98-043 | IPN 98-043 | Instrument Channel Surveillance Intervals Extended to 24 Months | Incorporated |
| T 4.1-2(1) | 139 | 139 | No TSCRs | No TSCRs for this Page | N/A |

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.1:
## "Accumulators"

| | | | | | |
|---|---|---|---|---|---|
| **4.5-7** | **178** | **178** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **4.5-8** | **178** | **178** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **4.5-9** | **148** | **148** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **4.5-11** | **148** | **148** | **No TSCRs** | No TSCRs for this Page | **N/A** |

SEE ITS 3.5.3 3.5.4

c. One residual heat removal pump and heat exchanger together with the associated piping and valves operable.

d. One recirculation pump together with its associated piping and valves operable.

2. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.1 within 1 hour the reactor shall be in the cold shutdown condition within the next 20 hours.

LCO 3.5.1 Applicability

3. ~~The reactor coolant system $T_{avg}$ shall not exceed 350°F unless the following requirements are met:~~ Mode 1, 2 & Mode 3 with prz >1000 A.5

SEE ITS 3.5.4

a. The refueling water storage tank water level shall be a minimum of 35.4 feet, with the water at a boron concentration ≥2400 ppm and ≤2600 ppm.

b. DELETED

LCO 3.5.1 c. The four accumulators are Operable ~~pressurized~~ between 600 and 700
SR 3.5.1.3 psig and each contains a minimum of 775 ft³ and a maximum
SR 3.5.1.2 of 815 ft³ of water at a boron concentration ≥2000 ppm and
SR 3.5.1.4 ≤2600 ppm. Accumulator isolation valves 894A, B, C, and D
SR 3.5.1.1 shall be open and their power supplies deenergized whenever
SR 3.5.1.5 the reactor coolant system pressure is ~~above~~ 1000 psig.

≥1000 psig A.5

SR 3.5.1.1 M.1

SR 3.5.1.5 M.1

≥2000 L.3

A.1

LA.1

d. ~~One pressure and one level transmitter shall be operating per accumulator.~~

SEE ITS 3.5.2

e. Three safety injection pumps together with their associated piping and valves are operable.

f. Two residual heat removal pumps and heat exchangers together with their associated piping and valves are operable.

g. Two recirculation pumps together with the associated piping and valves are operable.

h. Valves 856B and 856G in the Safety Injection discharge headers shall be closed and their power supplies de-energized.

i. Valve 1810 in the suction line of the high-level SI pumps and valves 882 and 744 in the suction and discharge lines, respectively, of the residual heat removal pumps shall be open and their power supplies de-energized.

j. Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the RWST are de-energized in the open position.

SEE ITS 3.3.3

k. The refueling water storage tank low level alarms are operable and set to alarm between 10.5 feet and 12.5 feet of water in the tank.

l. Valve 883 in the RHR return line to the RWST is de-energized in the closed position.

m. Valves 1870 and 743 in the miniflow line for the Residual Heat Removal Pumps shall be open and their power supplies de-energized.

n. The RHR system is in the ESF alignment with the normal RHR suction line isolated from the RCS.

LCO 3.5.1 Actions

4. The requirements of 3.3.A.3 may be modified to allow ~~any one of~~ the following components to be inoperable at ~~any one time~~ L.4

A.1

In Mode 3 — for up to 8 hours — M.6

Notes to LCO 3.5.1

~~a.~~ The accumulators may be isolated during the performance of the reactor coolant system hydrostatic tests.

For the purpose of accumulator check valve leakage testing, one accumulator may be isolated at a time, for up to 8 hours, provided the reactor is in ~~the hot shutdown condition~~ Mode 3

SEE ITS 3.5.2

b. One safety injection pump may be out of service, provided the pump is restored to an operable status within 24 hours.

c. One residual heat removal pump may be out of service, provided the pump is restored to an operable status within 24 hours.

d. One residual heat exchanger may be out of service provided that it is restored to an operable status within 48 hours.

e. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided that it is restored to an operable status within 24 hours and all valves in the system that provide the duplicate function are operable.

f. DELETED

SEE ITS 3.3.3

g. One refueling water storage tank low level alarm may be inoperable for up to 7 days provided the other low level alarm is operable.

Add Condition A and associated Req. Act (L.2)



Add Condition B and associated Req. Act (L.1)

LCO 3.5.1 ~~5.~~ ACTIONS

~~If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.3 within the time periods specified in 3.3.A.4, then:~~

Req. Act. C.1, C.2

a. ~~If~~ the reactor ~~is critical, it~~ shall be in the hot shutdown condition within four hours and the cold shutdown condition within the following 24 hours. (M.3) (M.4) (L) (L.1)

b. ~~If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.A.3 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition using normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.~~ (M.3)

SEE ITS 3.4.6 3.4.7 3.4.8

6. When the reactor coolant system $T_{avg}$ is greater than 200°F and less than 350°F, the following decay heat removal requirements shall be met:

a. Two residual heat removal pumps together with their associated heat exchangers, piping, and valves shall be operable,

**OR**

b. A minimum of one residual heat removal pump and heat exchanger and a minimum of one reactor coolant pump and steam generator together with their associated piping and valves, shall be operable,

**OR**

c. A minimum of two reactor coolant pumps and two steam generators, together with their associated piping and valves, shall be operable,

**OR**

d. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system. Otherwise, if sufficient equipment is available, be in cold shutdown within 20 hours.

7. When the reactor coolant $T_{avg}$ is less than 200°F, but not in the refueling operation condition, two residual heat removal pumps, together with their associated heat exchangers, piping and valves, shall be operable.

a. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system.

b. The above requirements may be suspended during maintenance, modifications, testing, inspection or repair provided that:

1) an alternate means of decay heat removal is available and return of the system within sufficient time to prevent exceeding cold shutdown requirements is assured;

Add Condition D and associated Req. Act (A.4)

Bases

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature, by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.[1] With this mode of startup, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation, and, therefore, the minimum required engineered safeguards and auxiliary cooling systems are required to be operable.

The probability of sustaining both a major accident and a simultaneous failure of a safeguards component to operate as designed is necessarily very small. Thus, operation with the reactor above the cold shutdown condition with minimum safeguards operable for a limited period does not significantly increase the probability of an accident having consequences which are more severe than the Design Basis Accident.

A.1

The operable status of the various systems and components is demonstrated by periodic tests defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible, in most cases, to effect repairs and restore the system to full operability within a relatively short time. The inoperability of a single component does not negate the ability of the system to perform its function,[2] but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. Assurance that the redundant component(s) will operate if required to do so exists if the required periodic surveillance testing is current and there are no known reasons to suggest that the redundant component(s) are inoperable. If it develops that (a) the inoperable component is not repaired within the specified allowable time period, or (b) a second component in the same or related system is found to be inoperable, the reactor, if critical, will initially be brought to the hot shutdown condition utilizing normal operating procedures to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. If the reactor was already subcritical, the reactor coolant system temperature and pressure will be maintained within the stated values in order to limit the amount of stored energy in the reactor coolant system. The stated tolerances provide a band for operator control. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the

cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would damage the fuel elements or result in a release in excess of 10 CFR 100 and 10 CFR 50 dose limits.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and, therefore, in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are based on two considerations:

1) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full rated power, the magnitude of the decay heat decreases after initiating hot shutdown. Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 1 hour of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case the reactor is to be put into the cold shutdown condition.

The limits for the Refueling Water Storage Tank and the accumulators insure the required amount of water with the proper boron concentration for injection into the reactor coolant system following a loss-of-coolant accident is available. These limits are based on values used in the accident analysis.(9)(13)

A.1

A.1

The minimum indicated RWST level of 35.4 feet (approximately 342,200 gals.), and the low level alarms ("allowable values") of 10.5 feet (approx. 111,100 gals.) and 12.5 feet (approx. 129,700 gals.) include consideration for instrumentation uncertainties, margin, and the unusable volume at the bottom of the tank.[17][18] These water levels ensure a minimum of approx. 195,800 gals. available for injection, and approx. 66,700 gals. for use during and following the transition from injection to recirculation (to allow continued CS pump operation for sump pH control).[18] The minimum RWST boron concentration ensures that the reactor core will remain subcritical during long term recirculation with all control rods fully withdrawn following a postulated large break LOCA.

The four accumulator isolation valves (894 A,B,C,D) are maintained in the open position when the reactor coolant pressure is above 1000 psig to assure flow passage from the accumulators will be available during the injection phases of a loss-of-coolant accident. Indication is also provided on the monitor light panel, should any of these valves not be in the full open position even with the valve operator deenergized. The 1000 psig limit is derived from the minimum pressure requirements of the accumulators combined with instrument error and an operational band and is based upon avoiding inadvertent injection into the reactor coolant system. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occurring when accumulator core cooling flow is required. Valves 856 B and G are maintained in the closed position to prevent hot leg injection during the injection phase of a loss-of-coolant accident. As an additional assurance of preventing hot leg injection, these valve motor operators are deenergized to prevent spurious opening of these valves during the injection phase of a loss-of-coolant accident. Power will be restored to these valves at an appropriate time in accordance with plant operating procedures after a loss-of-coolant accident in order to establish hot leg recirculation.

Valves 1810, 882, and 744 are maintained in the open position to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As additional assurance of flow passage availability, these valve motor operators are de-energized to prevent an extremely unlikely spurious closure. This additional precaution is acceptable, since failure to manually re-establish power to close these valves following the injection phase is tolerable as a single failure.

Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the refueling water storage tank are de-energized in the open position to prevent an extremely unlikely spurious closure which would cause the safety injection pumps to overheat if the reactor coolant system pressure is above the shutoff head of the pumps.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.[3] The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of their performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met.[13]

During operating modes in the temperature range between 200°F and 350°F, a sufficient decay heat removal capability is provided by a reactor coolant pump with a steam generator heat sink or a residual heat removal loop. This redundancy ensures that a single failure will not result in a complete loss of decay heat removal. Above 350°F, the normal RHR suction line is isolated from the RCS to protect RHR piping from overpressurization due to inadvertent SI pump actuation.

During operating modes when the reactor coolant $T_{avg}$ is less than 200°F, but not in the refueling operation condition, a sufficient decay heat removal capability is provided by a residual heat removal loop.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) containment spray with sodium hydroxide addition. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 95°F).[4] In the event of a Design Basis Accident, any one of the following configurations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting off-site doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also in the event of a Design Basis Accident, any one of three configurations of fan-cooler units (with charcoal filters) and/or containment spray pumps (with sodium hydroxide addition) will reduce airborne organic and molecular iodine activities sufficiently to limit off-site doses to acceptable values.[5] Any one of these three configurations constitutes the minimum safeguards for iodine removal.

The combination of three fan-coolers and one containment spray pump is capable of being operated on emergency power with one diesel generator failing to start. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of off-site power or operation of all emergency diesel generators.

**TABLE 4.1-1** (Sheet 3 of 6)

| Channel Description | Check | Calibrate | Test | Remarks |
|---|---|---|---|---|
| e. Main Steam Lines Process Radiation Monitors (R-62A, R-62B, R-62C, and R-62D) | D | 24M | Q | |
| f. Gross Failed Fuel Detectors (R-63A and R-63B) | D | 24M | Q | |
| 16. Containment Water Level Monitoring System: | | | | |
| a. Containment Sump | N.A. | 24M | N.A. | Narrow Range, Analog |
| b. Recirculation Sump | N.A. | 24M | N.A. | Narrow Range, Analog |
| c. Containment Water Level | N.A. | 24M | N.A. | Wide Range |
| 17. Accumulator Level and Pressure | S 12 hours | 18M*** | N.A. | |
| 18. Steam Line Pressure | S | 24M | Q | |
| 19. Turbine First Stage Pressure | S | 24M | Q | |
| 20a. Reactor Trip Relay Logic | N.A. | N.A. | TM | |
| 20b. ESF Actuation Relay Logic | N.A. | N.A. | TM | |
| 21. Turbine Trip Low Auto Stop Oil Pressure | N.A. | 24M | N.A. | |
| 22. DELETED | DELETED | DELETED | DELETED | |
| 23. Temperature Sensor in Auxiliary Boiler Feedwater Pump Building | N.A. | N.A. | 18M | |
| 24. Temperature Sensors in Primary Auxiliary Building | | | | |
| a. Piping Penetration Area | N.A. | N.A. | 24M | |
| b. Mini-Containment Area | N.A. | N.A. | 24M | |
| c. Steam Generator Blowdown Heat Exchanger Room | N.A. | N.A. | 24M | |

SR 3.5.1.2
SR 3.5.1.3

A.3

LA.1

Changed to 24 months by TSCR 98-043



TSCR 98-043

ITS 3.5.1

Deleted by TSCR 98-043

**Table Notation**

| | |
|---|---|
| * | By means of the movable incore detector system |
| ** | Quarterly when reactor power is below the setpoint and prior to each startup if not done previous month. |
| *** | ~~This surveillance requirement may be extended on a one time basis to no later than April 26, 1997.~~ |
| **** | ~~This surveillance requirement may be extended on a one time basis to no later than May 12, 1997.~~ |
| ***** | ~~This surveillance requirement may be extended on a one time basis to no later than May 14, 1997.~~ |
| # | These requirements are applicable when specification 3.3.F.5 is in effect only. |
| ## | The "each shift" frequency also requires verification that the DNB parameters (Reactor Coolant Temperature, Reactor Coolant Flow, and Pressurizer Pressure) are within the limits of Technical Specification 3.1.H. |

S - Each Shift (i.e., at least once per 12 hours)
W - Weekly
P - Prior to each startup if not done previous week
M - Monthly
NA - Not Applicable
Q - Quarterly
D - Daily
18M - At least once per 18 months
TM - At least every two months on a staggered test basis (i.e., one train per month)
24M - At least once per 24 months
6M - At least once per 6 months



TSCR 98-043

TABLE 4.1-2 (Sheet 1 of 2)

| FREQUENCIES FOR SAMPLING TESTS | | | |
|---|---|---|---|
| **Sample** | **Analysis** | **Frequency** | **Maximum Time Between Analysis** |
| 1. Reactor Coolant | Gross Activity[1]<br>Tritium Activity<br>Boron concentration<br>Radiochemical (gamma)[2] Spectral Check<br>Oxygen and Chlorides Concentration<br>Fluorides Concentration<br>$\bar{E}$ Determination [3]<br>Isotopic Analysis for I-131, I-133, I-135 | 5 days/week[1][4]<br>Weekly[1]<br>2 days/week<br>Monthly<br>3 times per 7 days<br>Weekly<br>Semi-Annually<br>Once per 14 days[3] | 3 days[4]<br>10 days<br>5 days<br>45 days<br>3 days<br>10 days<br>30 weeks<br>20 days |
| 2. Boric Acid Tank | Boron Concentration, Chlorides | Weekly | 10 days |
| 3. Spray Additive Tank | NaOH Concentration | Monthly | 45 days |
| 4. Accumulators | Boron Concentration | Monthly — 31 days | 45 days |
| 5. Refueling Water Storage Tank | Boron Concentration pH, Chlorides<br>Gross Activity | Monthly<br>Quarterly | 45 days<br>16 weeks |
| 6. Secondary Coolant | I-131 Equivalent (Isotopic Analysis)<br>Gross Activity | Monthly<br>3 times per 7 days | 45 days<br>3 days |
| 7. Component Cooling Water | Gross Activity, Corrosion Inhibitor and pH | Monthly | 45 days |
| 8. Spent Fuel Pool (when fuel stored) | Gross Activity Boron Concentration, Chlorides | Monthly | 45 days |

SR3.5.1.4

M.2

Amendment No. 139

Add SR3.5.1.4 accelerated Frequency

M.5

ITS 3.5.1

B. Component Tests

1. Pumps

SEE ITS 3.5.2 and ITS 3.6.6

a. The safety injection pumps, residual heat removal pumps, containment spray pumps and the auxiliary component cooling water pumps shall be started at quarterly intervals. The recirculation pumps shall be started at least once per 24 months.

b. Acceptable levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least fifteen minutes.

2. Valves

SEE ITS 3.6.7

a. Each spray additive valve shall be cycled by operator action with the pumps shut down at least once per 24 months.

LCO 3.5.1

b. ~~The accumulator check valves shall be checked for operability at least once per 24 months.~~ LA.2

SEE ITS 3.4.14

c. The following check valves shall be checked for gross leakage at least once per 24 months:

| | | | |
|---|---|---|---|
| 857A & G | 857J | 857S & T | 897B |
| 857B | 857K | 857U & W | 897C |
| 857C | 857L | 895A | 897D |
| 857D | 857M | 895B | 838A |
| 857E | 857N | 895C | 838B |
| 857F | 857P | 895D | 838C |
| 857H | 857Q & R | 897A | 838D |



Amendment No. ~~125~~, ~~129~~, ~~148~~, 178

d. In addition to 4.5.B.2.c, the following check valves shall be checked for gross leakage every time the plant is shut down and the reactor coolant system has been depressurized to 700 psig or less. This gross leakage test shall also be performed following valve maintenance, repair or other work which could unseat these check valves:

SEE ITS 3.4.14

| | | |
|---|---|---|
| 838A | 895A | 897A |
| 838B | 895B | 897B |
| 838C | 895C | 897C |
| 838D | 895D | 897D |

Basis

A.1

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally on standby during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes reactor trip, main feedwater isolation and containment isolation, and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is, therefore, to combine systems tests to be performed during plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action and verification made that the components receive the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[1]

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly (in accordance with Specification 4.1). The testing of the analog channel inputs is accomplished in the same manner as for the reactor protection system. The engineered safety features logic system is tested by means of test switches to simulate inputs from the analog channels. The test switches allow actuation of the master relay, while at the same time blocking the slave relays. Verification that the logic is accomplished is indicated by the matrix test light. The slave relay coil circuits are continuously verified by a built-in monitoring circuit. In addition, the active components (pumps and valves) are to be tested in accordance with the Indian Point 3 Inservice Testing Program. The pumps, specified in the Technical Specifications, are tested on a quarterly basis to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The exception to this quarterly test are the recirculation pumps which are tested during a refueling outage. The quarterly test interval is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent testing would result in increased wear over a long period of time.

Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System, and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification A.1, the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation, and by these means are continuously monitored for satisfactory performance.

The charcoal portion of the containment air recirculation system is a passive safeguard which is isolated from the cooling air flow during normal reactor operation. Hence, the charcoal should have a long useful lifetime. The filter frames that house the charcoal are stainless steel and should also last indefinitely. However, the visual inspection specified in Section A.4(a) of this specification will be performed to verify that this is, in fact, the case. The iodine removal efficiency cannot be measured with the filter cells in place. Therefore, at periodic intervals a representative sample of charcoal is to be removed and tested to verify that the efficiencies for removal of methyl iodide are obtained.(2) The fuel storage building air treatment system is designed to filter the discharge of the fuel storage building atmosphere to the facility vent during normal conditions. As required by Specifications 3.8.A.12 and 3.8.C.6, the fuel storage building emergency ventilation system must be operable whenever irradiated fuel is being moved. However, if the irradiated fuel has had a continuous 45-day decay period, the fuel storage building emergency ventilation system is not technically necessary, even though the system is required to be operable during all fuel handling operations. The emergency ventilation fan is automatically started upon high radiation signal and since the bypass assembly is sealed by manually operated isolation devices, air flow is directed through the emergency ventilation HEPA filters and charcoal adsorbers.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of these adsorbers for all emergency air treatment systems. The charcoal adsorbers are installed to reduce the potential release of radio-iodine to the environment. The in-place test results should indicate a system leak tightness of less than or equal to one percent leakage for the charcoal adsorbers and a HEPA efficiency of greater than or equal to 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of greater than or equal to 90 percent on the fuel handling system samples, and greater than or equal to 85 percent on the containment system samples for expected accident conditions. With the efficiencies of the HEPA filters and charcoal adsorbers as specified, further assurance is provided that the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed.

The basis for the toxic gas monitoring system is given in Technical Specification Section 3.3.

The control room air treatment system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room air treatment system is designed to automatically start upon control room isolation.

(A.1)

For the eight flow distribution valves (856 A, C, D, E, F, H, J and K), verification of the valve mechanical stop adjustments is performed periodically to provide assurance that the high head safety injection flow distribution is in accordance with flow values assumed in the core cooling analysis.

Gross leakage testing of the reactor coolant system pressure isolation valves and the Low Pressure Injection(LPI)/residual heat removal(RHR)system valves reduces the probability of an inter-system LOCA[4]. These tests implement the requirements set forth in NRC generic letter dated February 23, 1980, regarding testing of LPI/RHR system check valves. This amendment provides a basis for the rescission of item A.5. of a Confirmatory Order issued by the Commission to Indian Point 3 in a letter dated February 11, 1980. To satisfy ALARA requirements, gross leakage (>10 gpm) may be measured indirectly (i.e. using installed pressure and flow indications).

References

(1) FSAR Section 6.2

(2) FSAR Section 6.4

(3) FSAR Section 6.8

(4) WASH 1400

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.1:
## "Accumulators"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

## DISCUSSION OF CHANGES
## ITS SECTION 3.5.1 - Accumulators

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 (Superceded by Amendment 181) CTS Table 4.1-1, Item 17, requires verification every shift that Emergency Core Cooling System (ECCS) accumulator level and pressure are within required limits. ITS SR 3.5.1.2 and ITS SR 3.5.1.3 require verification that ECCS accumulator borated water volume and nitrogen cover pressure are within required limits every 12 hours.

This is an administrative change with no impact on safety because of the following: a 12 hour Frequency for verification of level and pressure is consistent with assumptions for the timely identification of a parameter outside required limits; these checks are normally supplemented by less formal checks of these parameters during normal operational use of the control displays associated with this LCO; operating experience demonstrates indicating instrument channel failure is rare; and, the length of a shift is not specified and could be either 8 or 12 hours.

A.4 CTS 3.3.A.5.a specifies that if requirements for ECCS accumulators are not met for one or more accumulators then the reactor must be in hot shutdown (Mode 3) within four hours and cold shutdown (Mode 5) within the following 24 hours. ITS 3.5.1, Conditions B and C, maintains this requirement to initiate a reactor shutdown within one hour if one accumulator is inoperable. However, ITS 3.5.1, Required Action D.1, is added to require that ITS LCO 3.0.3 be entered if two or more accumulators are inoperable (i.e., Mode 3 within 7 hours and less than 1000 psig reactor system pressure in less than 13 hours).

This change is needed because four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that there is no requirement to assume a random failure of an accumulator because it is a passive component and the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 could be violated. Therefore, even the one hour allowed to restore one inoperable accumulator provided in ITS 3.5.1, Required Action B.1, is not appropriate.

The change in the ITS 3.5.1, Required Action D.1, completion time to reach Mode 3 in 7 hours (versus 4 hours in CTS 3.3.A.5.a) is equivalent to the change introduced by ITS 3.5.1, Required Action B.1 (See ITS 3.5.1, DOC L.1). The change in the ITS 3.5.1, Required Action D.1 (less than 1000 psig reactor system pressure in less than 13 hours), is equivalent to the change introduced by ITS 3.5.1, Required Action C.2 (See ITS 3.5.1, DOC M.4). Therefore, adding a specific requirement to enter LCO 3.0.3 and place the plant in a condition in which the accumulators are not required whenever the plant is outside the accident analysis is an administrative change with no significant adverse impact on safety.

A.5 CTS 3.3.A.3 requires ECCS accumulators Operable when above 350 °F (Mode 3); however, CTS 3.3.A.3.c requires accumulator isolation valves open only when reactor coolant system pressure is above 1000 psig. Therefore, the CTS Applicability for ECCS accumulators is reactor coolant system temperature above 350 °F (Mode 3) and reactor coolant system pressure above 1000 psig. ITS LCO 3.5.1 maintains this Applicability. The improved presentation of Applicability requirements for ECCS accumulators is an administrative change with no impact on safety.

MORE RESTRICTIVE

M.1 CTS 3.3.A.3.c requires accumulator isolation valves 894A, 894B, 894C, and 894D open and their power supplies de-energized whenever the reactor coolant system pressure is above 1000 psig.

ITS SR 3.5.1.1 and 3.5.1.5 are added to require verification that each accumulator isolation valve is fully open every 12 hours and that power is removed from each isolation valve every 31 days (See ITS 3.5.1, DOC L.3).

This change is needed to require periodic verification that the requirements for ECCS accumulator Operability are met. These surveillances ensure that accumulators are available for injection by ensuring timely discovery if a valve is less than fully open and by ensuring that a failure does not result in the undetected closure of an accumulator motor operated isolation valve. This change is acceptable because it does not introduce any operation that is un-analyzed while requiring periodic verification that analysis assumptions regarding the Operability of ECCS accumulators are satisfied. Therefore, this change has no significant adverse impact on safety.

M.2 CTS Table 4.1-2, Item 4, specifies that the Frequency for verification of accumulator boron concentration is 31 days and that the maximum time between accumulator boron concentration verification should never exceed 45 days. ITS SR 3.5.1.4 maintains the requirement to verify boron concentration in each accumulator every 31 days; however, ITS SR 3.0.2 limits any extension to the 31 day SR interval to 25% (approx. 39 days).

This change is needed to establish consistent allowances for extending SR Frequencies consistent with ITS SR 3.0.2. This change is acceptable because it does not introduce any operation that is un-analyzed while requiring more timely verification that analysis assumptions regarding the Operability of ECCS accumulators are satisfied. Therefore, this change has no significant adverse impact on safety.

M.3 CTS 3.3.A.5 establishes the Actions required if the ECCS systems (Refueling Water Storage Tank, Accumulators, Safety Injection Pumps, Residual Heat Removal Pumps, Recirculation Pumps) are not restored to meet CTS requirements within specified completion times when above 350 °F (Mode 3). CTS 3.3.A.5.a specifies that, if the reactor is critical when requirements are not met, then the reactor must be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within the following 24 hours. However, if the reactor is subcritical when requirements are not met, CTS 3.3.A.5.b requires only that reactor coolant system temperature and pressure not be increased more than 25°F and 100 psi, respectively, over existing values with the requirement to proceed to cold shutdown (Mode 5) deferred by 48 hours.

Under the same conditions, ITS 3.5.1, Required Actions C.1 and C.2, require that the reactor be in Mode 3 in 6 hours (See ITS 3.5.1, DOC L.1) and reactor coolant system pressure reduced to less than 1000 psig within 12 hours (See ITS 3.5.1, DOCs M.4) regardless of the status of the unit when the Condition is identified. The allowance provided in CTS 3.3.A.5.b is deleted.

This change is needed to eliminate the ambiguity created by CTS 3.3.A.5.b when performing a reactor shutdown and cooldown required by CTS 3.3.A.5 and to ensure that the plant is placed outside the LCO Applicability promptly when the LCO requirements are not met. This change is acceptable because placing the plant outside the LCO Applicability when LCO requirements are not met is conservative and there is no change in the CTS 3.3.A.5 requirement. This change has no significant adverse impact on safety.

M.4 CTS 3.3.A.5 establishes the Actions required if the ECCS systems (Refueling Water Storage Tank, Accumulators, Safety Injection Pumps, Residual Heat Removal Pumps, Recirculation Pumps) are not restored to meet CTS requirements within specified completion times when above 350 °F (Mode 3). CTS 3.3.A.5.a specifies that, if the reactor is critical when requirements are not met, then the reactor must be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within the following 24 hours.

Under the same conditions, ITS 3.5.1, Required Actions C.1 and C.2, require that the reactor be in Mode 3 in 6 hours (See ITS 3.5.1, DOC L.1) and reactor coolant system pressure reduced to less than 1000 psig within 12 hours. This is a more restrictive change because ITS 3.5.1, Required Action C.2, places the plant outside of the LCO Applicability within 12 hours whereas CTS 3.3.A.5.a could allow the plant to stay within the LCO Applicability for a period approaching 24 hours.

This change is needed to make the Required Actions for an inoperable accumulator consistent with the LCO Applicability for ECCS accumulators (See ITS 3.5.1, DOC A.5) and to ensure that the plant is placed outside the LCO Applicability promptly when the LCO requirements are not met. This change is acceptable because 12 hours is a reasonable time to reach the required plant conditions ($\leq$ 1000 psig) from full power conditions in an orderly manner and without challenging plant systems. Therefore, this change has no significant adverse impact on safety.

M.5 CTS Table 4.1-2, Item 4, specifies that the Frequency for verification of accumulator boron concentration is 31 days.

ITS SR 3.5.1.4 maintains the requirement to verify boron concentration in each accumulator every 31 days; however, ITS SR 3.5.1.4 includes a requirement to verify boron concentration within 6 hours in any accumulator that experiences a level increase greater than 10% of indicated level (approximately 8.4 cubic feet) that is not the result of addition from the RWST.

This change is needed because it provides prompt verification that inleakage has not caused a reduction in boron concentration to below the required limit. This change is acceptable because inleakage of 8.4 cubic feet of pure water would result in a boron concentration reduction of approximately 1% based on the nominal accumulator volume of 795 cubic feet of water. An increase in the accumulator volume of 3 cubic feet causes a change of approximately 10% in the indicated accumulator level. This change is consistent with the recommendation of NUREG-1366. This change does not introduce any operation that is un-analyzed while requiring more timely verification that analysis assumptions regarding the Operability of ECCS accumulators are satisfied if the accumulators are subject to inleakage. Therefore, this change has no significant adverse impact on safety.

M.6 CTS 3.3.A.4.a provides the allowance that ECCS accumulators may be isolated during performance of the reactor coolant system hydrostatic tests. This allowance permits the plant to perform reactor coolant system hydrostatic testing above 350 °F with all accumulators isolated. This is necessary because limits imposed by CTS Figure 4.3-1, Pressure/Temperature Limits for Hydrostatic Leak Test, could, in some instances, require reactor coolant system hydrostatic testing above 350 °F (Mode 3) at some point during the test.

ITS LCO 3.5.1, Note 1, maintains the allowance that permits the plant to perform reactor coolant system hydrostatic testing above 350 °F with all accumulators isolated; however, ITS LCO 3.5.1, Note 1, limits the duration of this allowance to 8 hours. This change is needed to ensure that the period of time that ECCS accumulators are isolated when ITS LCO 3.5.1 is applicable is limited to that period of time needed to perform required testing. This change is acceptable because it does not introduce any operation that is un-analyzed while limiting the time that ECCS accumulators are isolated when ITS LCO 3.5.1 is applicable to that period of time needed to perform required testing. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.3.A.5 establishes Actions required if ECCS systems (Accumulators, Refueling Water Storage Tank, Safety Injection Pumps, Residual Heat Removal Pumps, Recirculation Pumps) are not restored to meet CTS requirements within specified completion times when above 350 °F (Mode 3). CTS 3.3.A.5.a specifies that, if the reactor is critical when requirements are not met, then the reactor must be in hot shutdown (Mode 3) within 4 hours. When an accumulator is inoperable, immediate entry into CTS 3.3.A.5 is required because CTS 3.3.A.4 does not provide any allowable out of service time for an inoperable accumulator (other than the AOT for testing in CTS 3.3.A.4.a).

Under the same conditions, ITS 3.5.1, Required Action B.1, allows 1 hour to attempt restoration before a reactor shutdown must be initiated; and, ITS 3.5.1, Required Action C.1, requires that the reactor be in Mode 3 in 6 hours. ITS 3.5.1, Required Actions B.1 and C.1, extend the time allowed to reach Mode 3 when one or more accumulators (See ITS 3.5.1, DOC A.4) are not Operable from 4 hours to 7 hours.

This change provided in Required Action B.1 is needed because attempting restoration of a single inoperable accumulator before subjecting the plant to a shutdown transient with fewer than the required number of ECCS accumulators is prudent.

The change provided in Required Action C.1 is needed because 6 hours is a reasonable time, based on operating experience, to reach the required plant conditions (Mode 3) from full power conditions in an orderly manner and without challenging plant systems.

These changes are acceptable because of the low probability of a DBA occurring during the additional 3 hours allowed to reach Mode 3. Therefore, this change has no significant adverse impact on safety.

L.2 CTS 3.3.A.5 does not differentiate between an ECCS accumulator that is inoperable because boron concentration is not within required limits and an ECCS accumulator that is inoperable for any other reason. Under the same conditions, ITS 3.5.1, Condition A and associated Required Action A.1, extends the allowable out of service time (AOT) for a single ECCS accumulator that is inoperable because boron concentration is not within required limits from immediate shutdown to 72 hours.

This change is acceptable because the loss of coolant accident (LOCA) analyses do not take credit for the boron concentration in the accumulators in the injection phase. The boron concentration of the accumulators is considered in the LOCA analyses during the recirculation phase only. The impact of a single accumulator below the minimum boron concentration limit will have no effect on available ECCS water and an insignificant effect on core subcriticality considering the total borated water volume available in the RWST during the recirculation phase. Also, boron concentration changes slowly relative to the required SR Frequency for verification of boron concentration; therefore, it is not likely that boron concentration will be significantly outside of required limits when the condition is identified. Finally, there is a low probability of a DBA occurring during the additional 72 hours allowed to restore boron concentration in a single inoperable ECCS accumulator. Therefore, this change has no significant adverse impact on safety.

L.3 CTS 3.3.A.3.c requires that accumulator isolation valves 894A, 894B, 894C, and 894D are open and power supplies de-energized whenever the reactor coolant system pressure is above 1000 psig.

ITS LCO 3.5.1 maintains the requirement that accumulator isolation valves 894A, 894B, 894C, and 894D must be open whenever the reactor coolant system pressure is above 1000 psig; however, ITS SR 3.5.1.5 requires that power be removed from each accumulator isolation valve operator only when reactor coolant system pressure is ≥ 2000 psig. This change allows a short term deferral during plant startups and shutdowns to perform an action (remove power) designed to reduce the potential for an inadvertent closure of an ECCS accumulator isolation valve.

This change is needed because it allows operational flexibility by avoiding unnecessary delays to manipulate breakers during plant startup and shutdown. This change is acceptable because of the following: the isolation valves will be open whenever RCS pressure is ≥ 1000 psig (i.e., ECCS accumulators are capable of performing the required safety function): the duration of transitions between system pressures of 1000 psig and 2000 psig are short; plant startups and shutdowns are conducted infrequently; and, the low probability of an inadvertent closure of an isolation valve and a DBA LOCA during the deferral period. Additionally, the safety injection signal provided to the valves will open a closed valve (if otherwise Operable) in the event of a LOCA. Therefore, this change has no significant adverse impact on safety.

L.4 CTS 3.3.A.4 limits the number of concurrent inoperable ECCS components (Refueling Water Storage Tank, Accumulators, High Head Safety Injection Pumps (HHSI), Residual Heat Removal (RHR) Pumps, Recirculation Pumps) by allowing "any one" of these ECCS components to be inoperable "at any one time." Therefore, in addition to the specific directions provided in CTS 3.3.A.4.a through CTS 3.3.A.4.g, CTS 3.3.A.4 does not permit concurrent inoperability of the RWST, Accumulators, HHSI Pumps, RHR Pumps, or Recirculation Pumps.

ITS LCO 3.5.1, ECCS Accumulators, ITS LCO 3.5.2, ECCS Systems-Operating, and ITS LCO 3.5.4, RWST, do not establish any restrictions on the concurrent inoperability of the RWST, Accumulators, High Head Safety Injection Pumps, Residual Heat Removal Pumps, and Recirculation Pumps.

This change is acceptable because the minimum complement of ECCS systems assumed available in the safety analysis consists of the following: all 4 accumulators, 2 of the 3 HHSI pumps, 1 of the 2 RHR pumps, and 1 of the 2 recirculation pumps. This minimum complement of ECCS systems is sufficient to mitigate any design basis accident. Additionally, each of these ECCS systems provides a different safety function; therefore, more than the minimum required number of pumps and/or accumulators for any of these systems does not provide significant compensation for an inoperable pump and/or accumulator in an ECCS system that provides a different safety function. Finally, the very short (one hour) allowable out of service time (AOT) when an ECCS accumulator or the RWST is inoperable due to a critical feature not within limits (See ITS 3.5.4, DOC L.2 and ITS 3.5.1, DOC L.2) is not affected by a concurrent inoperability of another ECCS system in either the CTS or the ITS. Therefore, elimination of the restriction in CTS 3.3.A.4 that prohibits concurrent inoperable ECCS systems has no significant adverse consequence and is deleted.

## REMOVED DETAIL

LA.1 CTS 3.3.A.3.d requires the operation of one pressure and one level transmitter per accumulator; and, CTS Table 4.1-1 requires calibration of these instruments every 18 months.

ITS 3.5.1 maintains the requirement that ECCS accumulator pressure and level must be maintained within required limits; however, the requirement for the operation of one pressure and one level transmitter per accumulator will be maintained in the Technical Requirements Manual (TRM). This change is acceptable because meeting the ITS 3.5.1 SRs requires at least one pressure and one level transmitter operating for each accumulator and that these instruments are calibrated. Therefore, maintaining the requirement in Technical Specifications that ECCS accumulator pressure and level must be verified within required limits every 12 hours and maintaining requirements for operation and calibration of instruments required to perform these verification in the TRM provides an adequate level of assurance that ECCS accumulators will be maintained within required limits.

The Quality Assurance Plan will be revised to specify that requirements in the TRM are part of the facility as described in the FSAR and that changes to the TRM can be made only in accordance with the requirements of 10 CFR 50.59. Therefore, this change is acceptable because there is no change to the existing requirements by the relocation of requirements to the TRM and future changes to the TRM will be controlled in accordance with 10 CFR 50.59.

This change is a less restrictive administrative change with no impact on safety because ITS 3.5.1 maintains the requirements to have ECCS accumulators Operable and maintains the requirements to perform periodic verification that demonstrates ECCS accumulator Operability. Therefore, requirements to operate and calibrate ECCS accumulator pressure and level instruments can be maintained in the TRM with no significant adverse impact on safety.

LA.2 CTS 4.5.B.2.b requires that the accumulator check valves be checked for Operability at least once per 24 months. ITS 3.5.1 maintains the requirement that ECCS accumulators must be Operable; however, the requirement to test the accumulator check valves is included in the Inservice Test (IST) Program, which provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The IST program is required by ITS 5.5.7.

ITS 5.5.7, Inservice Testing Program (IST), requires establishing and maintaining a program for inservice testing of ASME Code Class 1, 2, and 3 components at frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code. Additionally, 10 CFR 50.55a(f) already provides the regulatory requirements for this IST Program, and specifies that ASME Code Class 1, 2, and 3 pumps and valves are covered by an IST Program. Therefore, maintaining the requirement that ECCS accumulators must be Operable in ITS 3.5.1 and maintaining the requirement for periodic testing of accumulator check valves in the IST Program required by ITS 5.5.7 provides a high degree of assurance that check valves will be tested and maintained to ensure ECCS accumulator Operability. Additionally, ITS 5.5.7, Inservice Testing Program (IST), requirements and 10 CFR 50.55a(f) ensure adequate change control and regulatory oversight for any changes to the existing requirements. Therefore, requirements to test ECCS accumulator check valves can be maintained in the IST program with no significant adverse impact on safety.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.5.1:
# "Accumulators"

## PART 4:

## No Significant Hazards Considerations
## for
## Changes between CTS and ITS
## that are
## Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.5.1 - Accumulators

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS 3.5.1, Required Actions B.1 and C.1, modifies CTS 3.3.A.5.a requirements to extend the time allowed to reach Mode 3 when one or more accumulators are not Operable from 4 hours to 7 hours.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the time allowed to reach Mode 3 when requirements for ECCS Accumulators are not met has no affect on the initiators of any analyzed events. Additionally, 1 hour to attempt restoration is prudent to attempt restoration of a single inoperable accumulator before subjecting the plant to a shutdown transient with fewer than the required number of ECCS accumulators. Furthermore, 7 hours is a reasonable time, based on operating experience, to reach the required plant conditions (Mode 3) from full power conditions in an orderly manner and without challenging plant systems.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because the initial conditions for the initiation of the accident, one or more accumulators not Operable, is not changed.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS accumulators are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the initial conditions for the initiation of the accident, one or more accumulators not Operable, is not changed, the low frequency for which this allowance is expected to be used, and the low probability of a DBA occurring during the additional 3 hours allowed to reach Mode 3.

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LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS 3.5.1, Condition A and associated Required Action A.1, modifies CTS 3.3.A.5.a requirements to extend the allowable out of service time (AOT) for a single ECCS accumulator that is inoperable because boron concentration is not within required limits from immediate shutdown to 72 hours.

   This change will not result in a significant increase in the probability of an accident previously evaluated because ECCS accumulator boron concentration has no affect on the initiators of any analyzed events.

This change will not result in a significant increase in the consequences of an accident previously evaluated because the loss of coolant accident (LOCA) analyses do not take credit for the boron concentration in the accumulators in the injection phase. The boron concentration of the accumulators is considered in the LOCA analyses during the recirculation phase. The impact of a single accumulator below the minimum boron concentration limit will have no effect on available ECCS water and an insignificant effect on core subcriticality considering the total borated water volume available during the recirculation phase. Also, if the boron reduction is outside the limits it will likely not be by a significant amount. Finally, there is a low probability of a DBA occurring during the additional 72 hours allowed to restore boron concentration in a single inoperable ECCS accumulator.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way boron concentration is verified. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the loss of coolant accident (LOCA) analyses do not take credit for the boron concentration in the accumulators in the injection phase. The boron concentration of the accumulators is considered in the LOCA analyses during the recirculation phase. The impact of a single accumulator below the minimum boron concentration limit will have no effect on available ECCS water and an insignificant effect on core subcriticality considering the total borated water volume available during the recirculation phase. Also, if the boron reduction is outside the limits it will likely not be by a significant amount. Finally, there is a low probability of a DBA occurring during the additional 72 hours allowed to restore boron concentration in a single inoperable ECCS accumulator.

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LESS RESTRICTIVE
("L.3" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS LCO 3.5.1 maintains the requirement in CTS 3.3.A.3.c that accumulator isolation valves must be open whenever the reactor coolant system pressure is above 1000 psig: however, ITS SR 3.5.1.5 defers the requirement that power is removed from each accumulator isolation valve operator until reactor coolant system pressure is ≥ 2000 psig.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the status of power to the ECCS accumulator isolation valves has no affect on the initiators of any analyzed events.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because of the following: the isolation valves will be open whenever reactor coolant system pressure is > 1000 psig (i.e., ECCS accumulators are capable of performing the required safety function); the duration of transitions between system pressures of 1000 psig and 2000 psig is relatively short; plant startups and shutdowns are conducted infrequently; and, the low probability of an inadvertent closure of an isolation valve and a DBA LOCA during the deferral period. Additionally, the safety injection signal provided to the valves will open a closed valve (if otherwise Operable) in the event of a LOCA.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS accumulator isolation valves are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because of the following: the isolation valves will be open whenever reactor coolant system pressure is ≥ 1000 psig (i.e., ECCS accumulators are capable of performing the required safety function); the duration of transitions between system pressures of 1000 psig and 2000 psig is relatively short; plant startups and shutdowns are conducted infrequently; and, the low probability of an inadvertent closure of an isolation valve and a DBA LOCA during the deferral period. Additionally, the safety injection signal provided to the valves will open a closed valve (if otherwise Operable) in the event of a LOCA.

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LESS RESTRICTIVE
("L.4" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS LCO 3.5.1, ECCS Accumulators, ITS LCO 3.5.2, ECCS Systems-Operating, and ITS LCO 3.5.4, RWST, eliminates the CTS 3.3.A.4 restrictions on the concurrent inoperability of the RWST, Accumulators, High Head Safety Injection Pumps, Residual Heat Removal Pumps, and Recirculation Pumps.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the Operability status of ECCS systems has no affect on the initiators of any analyzed events.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because all 4 accumulators, 2 of the 3 HHSI pumps, 1 of the 2 RHR pumps, and 1 of the 2 recirculation pumps is the minimum complement of ECCS systems assumed available in the safety analysis and this minimum complement is sufficient to mitigate a design basis. Additionally, each of these ECCS systems provides a different safety function; therefore, more than the minimum required number of pumps and/or accumulators for any of these systems does not provide significant compensation for an inoperable pump and/or accumulator in an ECCS system that provides a different safety function. Finally, the very short (one hour) allowable out of service time (AOT) when an ECCS accumulator or the RWST is inoperable due to a critical feature not within limits is not affected by a concurrent inoperability of another ECCS system in either the CTS or the ITS.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS accumulator isolation valves are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because all 4 accumulators, 2 of the 3 HHSI pumps, 1 of the 2 RHR pumps, and 1 of the 2 recirculation pumps is the minimum complement of ECCS systems assumed available in the safety analysis and this minimum complement is sufficient to mitigate a design basis. Additionally, each of these ECCS systems provides a different safety function; therefore, more than the minimum required number of pumps and/or accumulators for any of these systems does not provide significant compensation for an inoperable pump and/or accumulator in an ECCS system that provides a different safety function. Finally, the very short (one hour) allowable out of service time (AOT) when an ECCS accumulator or the RWST is inoperable due to a critical feature not within limits is not affected by a concurrent inoperability of another ECCS system in either the CTS or the ITS.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.5.1:
# "Accumulators"

# PART 5:

# NUREG-1431
# Annotated to show differences between
# NUREG-1431 and ITS

## Status of NUREG 1431 Generic Changes for ITS 3.5.1

This ITS Specification is based on NUREG-1431 Specification No. **3.5.1**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| BWOG-025 | **155 R0** | 3.5.1 CORE FLOOD TANKS-DELETION OF CONDITION D AND MODIFICATION OF CONDITION C | Rejected by TSTF | Not Incorporated | N/A |
| WOG-057 | **117 R0** | REVISE ACCUMULATOR PRESSURE REFERENCE FROM PRESSURIZER TO RCS | Approved by NRC | Incorporated. TSTF is CLB. | T.1 |
| WOG-097 | | REVISE ACCUMULATOR BORON CONCENTRATION VERIFICATION SR | TSTF Review | Not Incorporated | N/A |

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

<CTS>

### 3.5.1 Accumulators

<3.3.A.3.c> LCO 3.5.1 [Four] ECCS accumulators shall be OPERABLE. Insert: 3.5-01-01 CLB.1

<3.3.A.3>
<3.3.A.3.c>
<DOC A.5>

APPLICABILITY: MODES 1 and 2,
MODE 3 with ~~pressurizer~~ Reactor coolant system pressure > [1000] psig. T.1

ACTIONS

| | CONDITION | REQUIRED ACTION | | COMPLETION TIME | |
|---|---|---|---|---|---|
| <DOC L.2> | A. One accumulator inoperable due to boron concentration not within limits of SR 3.5.1.4. | A.1 | Restore boron concentration to within limits of SR 3.5.1.4. | 72 hours | PA.1 |
| <3.3.A.5a> <DOC L.1> | B. One accumulator inoperable for reasons other than Condition A. | B.1 | Restore accumulator to OPERABLE status. | 1 hour | |
| <3.3.A.5.a> <DOC M.3> <DOC L.1> <DOC M.4> | C. Required Action and associated Completion Time of Condition A or B not met. | C.1<br><br>AND<br><br>C.2 | Be in MODE 3.<br><br><br><br>Reduce ~~pressurizer~~ Reactor Coolant System pressure to ≤ [1000] psig. | 6 hours<br><br><br><br>12 hours | T.1 |
| <DOC A.4> | D. Two or more accumulators inoperable. | D.1 | Enter LCO 3.0.3. | Immediately | |

NUREG-1431 Markup Inserts
ITS SECTION 3.5.1 - Accumulators

INSERT 3.5-1-01:

<CTS>

-----------------------------NOTES-----------------------------

<3.3.A.4.a> <DOC M.6> 1. In Mode 3, all accumulator discharge isolation valves may be closed and energized for up to 8 hours during the performance of reactor coolant system hydrostatic testing. CLB.1

<3.3.A.4.a> 2. In MODE 3, one accumulator discharge isolation valve be closed and energized for up to 8 hours for accumulator check valve leakage testing.

---------------------------------------------------------------

SURVEILLANCE REQUIREMENTS

| | | SURVEILLANCE | FREQUENCY |
|---|---|---|---|
| <3.3.A.3.c> <DOC M.1> | SR 3.5.1.1 | Verify each accumulator {discharge} isolation valve is fully open. | 12 hours |
| <3.3.A.3.c> <T 4.1-1, #17> | SR 3.5.1.2 | Verify borated water volume in each accumulator is ≥ ~~[2853 gallons (/)%]~~ {775 cubic feet} and ≤ ~~[8171 gallons (/)%]~~ {815 cubic feet}. | 12 hours DB.1 |
| <3.3.A.3.c> <T 4.1-1, #17> | SR 3.5.1.3 | Verify nitrogen cover pressure in each accumulator is ≥ ~~[385]~~ {600} psig and ≤ ~~[481]~~ {700} psig. | 12 hours |
| <3.3.A.3.c> <T 4.1.2, #4> <DOC M.5> <DOC M.2> | SR 3.5.1.4 | Verify boron concentration in each accumulator is ≥ ~~[1900]~~ {2000} ppm and ≤ ~~[2100]~~ {2600} ppm. | 31 days<br>AND<br>-----NOTE------<br>Only required to be performed for affected accumulators<br>--------------<br>Once within 6 hours after each solution volume increase of ≥ ~~[[ ] gallons, (/)% of indicated level]~~ 10% that is not the result of addition from the refueling water storage tank |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

<3.3.A.3.c>
<DOC M.1>
<DOC L.3>

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| SR 3.5.1.5 | Verify power is removed from each accumulator isolation valve operator when pressurizer pressure is ≥ [2000] psig. | 31 days |

Reactor coolant system

T.1

## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for ~~a small break LOCA~~.

Insert: B 3.5-1-01

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series. ~~The motor operated isolation valves are interlocked by P-11 with the pressurizer pressure measurement channels to ensure that the valves will automatically open as RCS pressure increases to above the permissive circuit P-11 setpoint.~~ DB.1

(continued)

DB.1

**INSERT B 3.5-1-01:**

any LOCA that reduces RCS pressure to below the accumulator pressure.

BASES

BACKGROUND (continued)

~~This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. The valves will automatically open, however, as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.~~

DB.1

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

(continued)

BASES

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APPLICABLE SAFETY ANALYSES (continued)

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and high head safety injection (HHSI) ~~centrifugal charging~~ pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the ~~centrifugal charging~~ HHSI pumps become solely responsible for terminating the temperature increase. DB.1

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. ~~3~~ 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. ~~For small breaks, an increase in water volume is a peak clad temperature penalty. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of [6468] gallons and [6879] gallons. To allow for instrument inaccuracy, values of [6520] gallons and [6820] gallons are specified.~~

Insert: B3.5-4-01

DB.1

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

injection of nitrogen into the RCS,

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. ~~2~~ 3 and 4).

The accumulators satisfy Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36.

PA.1

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.5.1 - Accumulators

DB.1

INSERT B 3.5-4-01:

Accumulator tank size and accumulator water volume directly affect the volume of nitrogen cover gas whose expansion produces the passive injection and thus affects injection rate. The amount of water is also important since the accumulator water which has not been injected and bypassed during blowdown is primarily responsible for filling the lower plenum (refill) and downcomer. The elevation head of the downcomer water provides the driving force for core reflooding (Ref. 3).

For large break LOCAs, changes in accumulator water volume can result in either improved or worsened analysis results; therefore, a nominal accumulator water volume of 795 cubic feet is modeled in the analysis (Ref. 3).

For small break LOCAs, changes in accumulator water volume are not significant because the clad temperature transient is terminated before the accumulators empty; therefore, a nominal accumulator water volume of 795 cubic feet is modeled in the analysis (Ref. 3).

BASES (continued)

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated. 2

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above ~~[~~2000~~]~~ psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 3) limit of 2200°F. 2

Insert: B3.5-5-01

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated discharge isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

CLB.1

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be

(continued)

INSERT B 3.5-5-01:

Note 1 provides an exception to SR 3.5.1.1 and SR 3.5.1.5 and specifies that all accumulator discharge isolation valves may be closed and energized for up to 8 hours during the performance of reactor coolant system hydrostatic testing. This allowance is necessary because limits imposed by the Pressure/Temperature Limits for a hydrostatic leak test, could, in some instances, require reactor coolant system hydrostatic testing above 350 °F (Mode 3). This allowance is acceptable because hydrostatic testing performed in MODE 3 when the need for the accumulators is reduced and Note 1 limits the duration the of time needed to perform required testing.

Note 2 also provides an exception to SR 3.5.1.1 and SR 3.5.1.5 and specifies that one accumulator discharge isolation valve may be closed and energized in MODE 3 for up to 8 hours for accumulator check valve leakage testing. This allowance is acceptable because testing is limited to MODE 3 when the need for the accumulators is reduced and Note 2 limits the duration to the time needed to perform required testing.

BASES

ACTIONS

A.1 (continued)

reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break ~~for the majority of plants~~. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits. PA.1

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and ~~pressurizer~~ pressure reduced to T.1

reactor coolant system

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

accumulator discharge

Each accumulator valve should be verified to be fully open every 12 hours. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling the affected accumulator within 6 hours after ~~a 1% volume~~ an increase of 3 cubic feet will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. ~~5~~ 4).

Insert: B 3.5-8-01

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator discharge isolation valve operator when the pressurizer pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated discharge isolation valves when pressurizer pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. ~~Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves.~~ DB.1

Should closure of a valve occur ~~in spite of the interlock~~, the SI signal provided to the valves would open a closed valve in the event of a LOCA. DB.1

(continued)

INSERT B 3.5-8-01:

Considering the nominal accumulator volume of 795 cubic feet of water, inleakage of 8.4 cubic feet of pure water would result in a boron concentration reduction of approximately 1%. An increase in the accumulator volume of 8.4 cubic feet causes a change of approximately 10%in the indicated accumulator level.

BASES (continued)

REFERENCES

~~1. IEEE Standard 279-1971.~~

1 ~~2~~. FSAR, Chapter [6].

2 ~~3~~. 10 CFR 50.46.

3 ~~4~~. FSAR, Chapter ~~[15]~~. 14

4 ~~5~~. NUREG-1366, February 1990.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.5.1: "Accumulators"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

## JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
## ITS SECTION 3.5.1 - Accumulators

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 IP3 ITS LCO 3.5.1 is modified by two notes: Note 1 allows the accumulator discharge isolation valves to be closed and energized for up to 8 hours during the performance of reactor coolant system hydrostatic testing; and, Note 2 allows one accumulator dischagre isolation valve to be closed and energized for up to 8 hours for accumulator check valve leakage testing. These changes maintain allowances provided in CTS 3.3.A.4.a (as modified by Discussion of Change M.6). These changes are needed and are acceptable because of the following:

Note 1 provides an exception to SR 3.5.1.1 and SR 3.5.1.5 and specifies that all accumulator discharge isolation valves may be closed and energized for up to 8 hours during the performance of the reactor coolant system hydrostatic testing. This allowance is necessary because limits imposed by the Pressure/Temperature Limits for a hydrostatic leak test, could, in some instances, require that reactor coolant system hydrostatic testing be performed above 350 °F (Mode 3). This allowance is acceptable because hydrostatic testing is performed very infrequently and would be performed in MODE 3 when the need for the accumulators is reduced. Additionally, Note 1 limits the duration to a period of time needed to perform required testing.

Note 2 also provides an exception to SR 3.5.1.1 and SR 3.5.1.5 and specifies that one accumulator discharge isolation valve may be closed and energized for up to 8 hours for accumulator check valve leakage testing performed in MODE 3. This allowance is acceptable because testing is performed infrequently and limited to MODE 3 when the need for the accumulators is reduced. Additionally, Note 2 limits the duration to a period of time needed to perform required testing.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

## PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

## DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-117, Rev.0 (WOG-57) which revises the reference pressure for the accumulator pressure limit from pressurizer pressure to reactor coolant system pressure. This change maintains the current licensing basis.

## DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.5.2:
# "ECCS - Operating"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

# 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## 3.5.2 ECCS – Operating

LCO 3.5.2 Three ECCS trains shall be OPERABLE.

---------------------------NOTES---------------------------

1. In MODE 3, both HHSI flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.

2. Operation in MODE 3 with HHSI pumps made incapable of injecting pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP)," is allowed for up to 4 hours or until the temperature of all RCS cold legs exceeds 375°F, whichever comes first.

-------------------------------------------------------------

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more trains inoperable.<br><br>AND<br><br>At least 100% of the ECCS flow equivalent to OPERABLE ECCS trains available. | A.1 Restore train(s) to OPERABLE status. | 72 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 4. | 6 hours<br><br><br><br>12 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.2.1 | Verify the following valves are in the listed position with power to the valve operator removed.<br><br>Number / Position / Function<br>SI-856B / Closed / HHSI Loop 33 Hot Leg Injection Stop Valve<br>SI-856G / Closed / HHSI Loop 31 Hot Leg Injection Stop Valve<br>SI-1810 / Open / RWST outlet isolation<br>AC-744 / Open / Common discharge isolation for RHR pumps<br>SI-882 / Open / Common RWST suction isolation for RHR pumps<br>SI-842 / Open / HHSI pump minimum flow line isolation<br>SI-843 / Open / HHSI pump minimum flow line isolation<br>SI-883 / Closed / RHR pump return to RWST isolation<br>AC-1870 / Open / RHR pump minimum flow line isolation<br>AC-743 / Open / RHR pump minimum flow line isolation | 12 hours |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.2.2 | Verify that each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. | 31 days |
| SR 3.5.2.3 | Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head. | In accordance with the Inservice Testing Program |
| SR 3.5.2.4 | Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. | 24 months |
| SR 3.5.2.5 | Verify each ECCS pump starts automatically on an actual or simulated actuation signal. | 24 months |
| SR 3.5.2.6 | Verify, for each ECCS throttle valve listed below, each position stop is in the correct position.<br><br><u>Valve Numbers</u><br><br>SI-856A SI-856F<br>SI-856C SI-856H<br>SI-856D SI-856J<br>SI-856E SI-856K | 24 months |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.2.7 | Verify, by visual inspection, each ECCS train containment sump suction inlet and recirculation sump suction inlet is not restricted by debris and the suction inlet screens show no evidence of structural distress or abnormal corrosion. | 24 months |

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.2 ECCS – Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the recirculation and containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the recirculation sump or containment sump for cold leg recirculation. After between 14.3 and 24 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS FUNCTION is provided by three separate ECCS systems: high head safety injection (HHSI), residual heat removal (RHR) injection, and containment recirculation. Each ECCS system is divided into subsystems as follows:

BASES

BACKGROUND (Continued)

- HHSI System is divided into three 50% capacity subsystems. Each HHSI subsystem consists of one pump as well as associated piping and valves to transfer water from the suction source to the core. HHSI subsystem 32 is OPERABLE when capable of injecting using the flow path associated with either HHSI subsystem 31 or 33. Note that the HHSI pumps have a shutoff head of approximately 1500 psig. Therefore, IP3 is classified as a low head safety injection plant.

- RHR injection System is divided into two 100% capacity subsystems. Each ECCS RHR subsystem consists of one RHR pump and one RHR heat exchanger as well as associated piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either RHR subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE RHR injection subsystem.

- Containment Recirculation is divided into two 100% capacity subsystems. Each subsystem consists of one Containment Recirculation pump and one RHR heat exchanger as well as associated piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either Recirculation subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE Containment Recirculation subsystem.

The three ECCS systems (3 HHSI, 2 RHR and 2 Recirculation) are grouped into three trains (5A, 2A/3A and 6A) such that any 2 of the 3 trains are capable of meeting all ECCS capability assumed in the accident analysis. Each ECCS train consists of the following:

a. ECCS Train 5A includes subsystems HHSI 31 and containment recirculation 31;

b. ECCS Train 2A/3A includes subsystems HHSI 32 and RHR 31; and,

BASES

BACKGROUND (Continued)

c. ECCS Train 6A includes subsystems HHSI 33, RHR 32, and containment recirculation 32.

The ECCS trains use the same designation as the Safeguards Power Trains required by LCO 3.8.9, Distribution Systems - Operating, with Safeguards Power Train 5A supported by DG 33, Safeguards Power Train 2A/23 supported by DG 31, Safeguards Power Train 6A supported by DG 32.

The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the high head safety injection pumps, the RHR pumps, heat exchangers, and the containment recirculation pumps. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from different trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the HHSI and RHR pumps. The discharge from the HHSI and RHR pumps feed injection lines to each of the RCS cold legs. Control valves are set to balance the HHSI flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

During the recirculation phase of LOCA recovery, the containment recirculation pumps take suction from the containment recirculation sump and direct flow through the RHR heat exchangers to the cold legs. The RHR pumps can be used to provide a backup method of recirculation in which case the RHR pump suction is transferred to the containment sump. The RHR pumps then supply recirculation flow directly or supply the suction of the HHSI pumps. Initially, recirculation is through

BASES

BACKGROUND (Continued)

the same paths as the injection phase. Subsequently, recirculation injection is split between the hot and cold legs.

The ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of HHSI pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems, exccept for the containment recirculation subsystems, are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The HHSI pumps are credited in a small break LOCA event. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one EDG; and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one EDG.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion

BASES

APPLICABLE SAFETY ANALYSES (continued)

for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the HHSI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36.

LCO

In MODES 1, 2, and 3, three ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting any one train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, the ECCS consists of the following:

a. ECCS Train 5A includes HHSI subsystem 31 and containment recirculation subsystem 31;

b. ECCS Train 2A/3A includes HHSI subsystem 32 and RHR subsystem 31; and,

c. ECCS Train 6A includes HHSI subsystem 33, RHR subsystem 32, and containment recirculation subsystem 32.

Each HHSI subsystem consists of one pump as well as associated instrumentation, piping and valves to transfer water from the suction source to the core. HHSI subsystem 32 is OPERABLE when capable of injecting using the flow path associated with either HHSI subsystem 31 or 33.

BASES

LCO (continued)

Each ECCS RHR subsystem consists of one RHR pump and one RHR heat exchanger as well as associated instrumentation, piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either RHR subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE RHR injection subsystem.

Each containment recirculation subsystem consists of one Containment Recirculation pump and one RHR heat exchanger as well as associated instrumentation piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either Recirculation subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE Containment Recirculation subsystem. Note that Recirculation pump OPERABILITY requires the functional availability of the associated auxiliary component cooling water pump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the HHSI and RHR pumps and their supply header to each of the four cold leg injection nozzles (8 cold leg injection nozzles for the HHSI pumps). In the long term, this flow path may be switched to take its supply from the containment recirculation sump using the containment recirculation pumps or, alternately, the containment sump using the RHR pumps to supply its flow to the RCS hot and cold legs, either directly into the RCS or via the HHSI pumps.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable more than one ECCS train (except as described in Reference 5).

As indicated in Note 1, the SI flow paths may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. This is acceptable because the flow paths are readily restorable from the control room or the valves are opened under administrative controls that ensure prompt closure when required. These administrative controls consist of stationing a dedicated

BASES

LCO (continued)

operator at the valve controls, who is in continuous communication with the control room.

As indicated in Note 2, operation in MODE 3 with ECCS trains made incapable of injecting pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be made incapable of injecting at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements when at lower power. The HHSI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

This LCO is only applicable in MODE 3 and above. Below MODE 3, system functional requirements are relaxed as described in LCO 3.5.3, "ECCS – Shutdown."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

BASES

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent (100% HHSI flow, 100% RHR injection flow, and 100% containment recirculation flow) to OPERABLE ECCS trains available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 4) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one pump in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different pumps, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to two OPERABLE ECCS trains remains available. This allows increased flexibility in plant operations under circumstances when pumps in redundant trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 4) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 5 describes situations in which one component, such as the valves governed by SR 3.5.2.1 and SR 3.5.2.6, can disable more than one ECCS train. With one or more component(s) inoperable such that 100% of the flow equivalent for HHSI, RHR and Containment Recirculation is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render more than one ECCS train inoperable. Securing these valves in position by removal of power or by key locking the control in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 5, that can disable the function of more than one ECCS train and invalidate the accident analyses. A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.2 (continued)

reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.4 and SR 3.5.2.5

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. Note that the Containment Recirculation system is a manually initiated system and is not included as part of this SR. Additionally, this Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.4 and SR 3.5.2.5 (continued)

under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.6

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves have stops to allow proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. Therefore, an improperly positioned valve could result in the inoperability of more than one injection flow path. The stops are set based on the results of the most recent ECCS operational flow test. The 24 month Frequency is based on the reasons stated in SR 3.5.2.4 and SR 3.5.2.5.

SR 3.5.2.7

Periodic inspections of each containment and recirculation sump suction inlet ensure that each is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency is sufficient to detect abnormal degradation and is confirmed by industry operating experience.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.

2. 10 CFR 50.46.

3. FSAR, Section 14.

4. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

5. IE Information Notice No. 87-01.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.2:
## "ECCS - Operating"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.3-2 | 154 | 154 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-3 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-4 | 139 | 139 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-5 | 53 | 53 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-14 | 132 | 132 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-15 | 139 TSCR 97-175 | 139 TSCR 97-175 | IPN 97-175 | Changes to Bases Pages | |
| 3.3-16 | 154 | 154 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-17 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-1 | 142 | 142 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-7 | 178 | 178 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-8 | 178 | 178 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-9 | 148 | 148 | No TSCRs | No TSCRs for this Page | N/A |

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.2:
## "ECCS - Operating"

| | | | | | |
|---|---|---|---|---|---|
| 4.5-11 | 148 | 148 | No TSCRs | No TSCRs for this Page | N/A |

A.1 A.2

SEE ITS 3.5.3 3.5.4

c. One residual heat removal pump and heat exchanger together with the associated piping and valves operable.

d. One recirculation pump together with its associated piping and valves operable.

2. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.1 within 1 hour the reactor shall be in the cold shutdown condition within the next 20 hours.

LCO 3.5.2 Applicability

3. ~~The reactor coolant system $T_{avg}$ shall not exceed 350°F unless the following requirements are met:~~ Mode 1, 2 and 3 — A.7

SEE ITS 3.5.4

a. The refueling water storage tank water level shall be a minimum of 35.4 feet, with the water at a boron concentration ≥2400 ppm and ≤2600 ppm.

b. **DELETED**

SEE ITS 3.5.1

c. The four accumulators are pressurized between 600 and 700 psig and each contains a minimum of 775 $ft^3$ and a maximum of 815 $ft^3$ of water at a boron concentration ≥2000 ppm and ≤2600 ppm. Accumulator isolation valves 894A, B, C, and D shall be open and their power supplies deenergized whenever the reactor coolant system pressure is above 1000 psig.

Add ITS 3.5.2, Applicability Note 1 — L.3

Add ITS 3.5.2, Applicability Note 2 — L.4

SEE ITS 3.5.1 — d. One pressure and one level transmitter shall be operating per accumulator.

LCO 3.5.2

~~e.~~ Three safety injection pumps together with their associated piping and valves are operable.

~~f.~~ Two residual heat removal pumps and heat exchangers together with their associated piping and valves are operable.

~~g.~~ Two recirculation pumps together with the associated piping and valves are operable.

LA.1

Verify every 12 hours — M.1

SR 3.5.2.1 — ~~h.~~ Valves 856B and 856G in the Safety Injection discharge headers shall be closed and their power supplies de-energized.

SR 3.5.2.1 — ~~i.~~ Valve 1810 in the suction line of the high-level SI pumps and valves 882 and 744 in the suction and discharge lines, respectively, of the residual heat removal pumps shall be open and their power supplies de-energized.

SR 3.5.2.1 — ~~j.~~ Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the RWST are de-energized in the open position.

SEE ITS 3.3.3 — k. The refueling water storage tank low level alarms are operable and set to alarm between 10.5 feet and 12.5 feet of water in the tank.

SR 3.5.2.1 — ~~l.~~ Valve 883 in the RHR return line to the RWST is de-energized in the closed position.

SR 3.5.2.1 — ~~m.~~ Valves 1870 and 743 in the miniflow line for the Residual Heat Removal Pumps shall be open and their power supplies de-energized.

~~n.~~ ~~The RHR system is in the ESF alignment with the normal RHR suction line isolated from the RCS.~~ — LA.4

LCO 3.5.2 Actions — 4. The requirements of 3.3.A.3 may be modified to allow ~~any one of~~ the following components to be inoperable ~~at any one time:~~ — L.1

Add SR 3.5.2.2 — M.1



Amendment No. ~~14~~, ~~154~~,179

SEE ITS 3.5.1

a. The accumulators may be isolated during the performance of the reactor coolant system hydrostatic tests.

For the purpose of accumulator check valve leakage testing, one accumulator may be isolated at a time, for up to 8 hours, provided the reactor is in the hot shutdown condition.

3.5.2 Req Act A.1

A.4 L.1

b. One safety injection pump may be out of service, provided the pump is restored to an operable status within 24 hours. 72 L2

c. One residual heat removal pump may be out of service, provided the pump is restored to an operable status within 24 hours. 72 L.2

d. One residual heat exchanger may be out of service provided that it is restored to an operable status within 48 hours. 72 L.2

e. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided that it is restored to an operable status within 24 hours and all valves in the system that provide the duplicate function are operable. 72 L.2

f. DELETED Recirculation subsys inoperable - 72 hours L.2

SEE ITS 3.3.3

g. One refueling water storage tank low level alarm may be inoperable for up to 7 days provided the other low level alarm is operable.

LCO 3.5.2 Actions (A.7)

5. ~~If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.3 within the time periods specified in 3.3.A.4; then:~~

a. ~~If~~ the reactor ~~is critical, it~~ shall be in the hot shutdown condition within four hours and the cold shutdown condition within the following 24 hours. (M.4) (M.3) 12 (L.6) 6

b. ~~If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.A.3 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition using normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.~~ (M.4)

SEE ITS 3.4.6 3.4.7 3.4.8

6. When the reactor coolant system $T_{avg}$ is greater than 200°F and less than 350°F, the following decay heat removal requirements shall be met:

a. Two residual heat removal pumps together with their associated heat exchangers, piping, and valves shall be operable,

**OR**

b. A minimum of one residual heat removal pump and heat exchanger and a minimum of one reactor coolant pump and steam generator together with their associated piping and valves, shall be operable,

**OR**

c. A minimum of two reactor coolant pumps and two steam generators, together with their associated piping and valves, shall be operable,

**OR**

d. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system. Otherwise, if sufficient equipment is available, be in cold shutdown within 20 hours.

7. When the reactor coolant $T_{avg}$ is less than 200°F, but not in the refueling operation condition, two residual heat removal pumps, together with their associated heat exchangers, piping and valves, shall be operable.

a. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system.

b. The above requirements may be suspended during maintenance, modifications, testing, inspection or repair provided that:

1) an alternate means of decay heat removal is available and return of the system within sufficient time to prevent exceeding cold shutdown requirements is assured;

**Bases**

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature, by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.[1] With this mode of startup, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation, and, therefore, the minimum required engineered safeguards and auxiliary cooling systems are required to be operable.

The probability of sustaining both a major accident and a simultaneous failure of a safeguards component to operate as designed is necessarily very small. Thus, operation with the reactor above the cold shutdown condition with minimum safeguards operable for a limited period does not significantly increase the probability of an accident having consequences which are more severe than the Design Basis Accident. A.1

The operable status of the various systems and components is demonstrated by periodic tests defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible, in most cases, to effect repairs and restore the system to full operability within a relatively short time. The inoperability of a single component does not negate the ability of the system to perform its function,[2] but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. Assurance that the redundant component(s) will operate if required to do so exists if the required periodic surveillance testing is current and there are no known reasons to suggest that the redundant component(s) are inoperable. If it develops that (a) the inoperable component is not repaired within the specified allowable time period, or (b) a second component in the same or related system is found to be inoperable, the reactor, if critical, will initially be brought to the hot shutdown condition utilizing normal operating procedures to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. If the reactor was already subcritical, the reactor coolant system temperature and pressure will be maintained within the stated values in order to limit the amount of stored energy in the reactor coolant system. The stated tolerances provide a band for operator control. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the

cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would damage the fuel elements or result in a release in excess of 10 CFR 100 and 10 CFR 50 dose limits.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and, therefore, in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are based on two considerations:

1) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full rated power, the magnitude of the decay heat decreases after initiating hot shutdown. Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 1 hour of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case the reactor is to be put into the cold shutdown condition.

The limits for the Refueling Water Storage Tank and the accumulators insure the required amount of water with the proper boron concentration for injection into the reactor coolant system following a loss-of-coolant accident is available. These limits are based on values used in the accident analysis.[9][13]

A.1

A.1

The minimum indicated RWST level of 35.4 feet (approximately 342,200 gals.), and the low level alarms ("allowable values") of 10.5 feet (approx. 111,100 gals.) and 12.5 feet (approx. 129,700 gals.), include consideration for instrumentation uncertainties, margin, and the unusable volume at the bottom of the tank.[17][18] These water levels ensure a minimum of approx. 195,800 gals. available for injection, and approx. 66,700 gals. for use during and following the transition from injection to recirculation (to allow continued CS pump operation for sump pH control).[18] The minimum RWST boron concentration ensures that the reactor core will remain subcritical during long term recirculation with all control rods fully withdrawn following a postulated large break LOCA.

The four accumulator isolation valves (894 A,B,C,D) are maintained in the open position when the reactor coolant pressure is above 1000 psig to assure flow passage from the accumulators will be available during the injection phases of a loss-of-coolant accident. Indication is also provided on the monitor light panel, should any of these valves not be in the full open position even with the valve operator deenergized. The 1000 psig limit is derived from the minimum pressure requirements of the accumulators combined with instrument error and an operational band and is based upon avoiding inadvertent injection into the reactor coolant system. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occurring when accumulator core cooling flow is required. Valves 856 B and G are maintained in the closed position to prevent hot leg injection during the injection phase of a loss-of-coolant accident. As an additional assurance of preventing hot leg injection, these valve motor operators are deenergized to prevent spurious opening of these valves during the injection phase of a loss-of-coolant accident. Power will be restored to these valves at an appropriate time in accordance with plant operating procedures after a loss-of-coolant accident in order to establish hot leg recirculation.

Valves 1810, 882, and 744 are maintained in the open position to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As additional assurance of flow passage availability, these valve motor operators are de-energized to prevent an extremely unlikely spurious closure. This additional precaution is acceptable, since failure to manually re-establish power to close these valves following the injection phase is tolerable as a single failure.

Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the refueling water storage tank are de-energized in the open position to prevent an extremely unlikely spurious closure which would cause the safety injection pumps to overheat if the reactor coolant system pressure is above the shutoff head of the pumps.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.[3] The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of their performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met.[13]

During operating modes in the temperature range between 200°F and 350°F, a sufficient decay heat removal capability is provided by a reactor coolant pump with a steam generator heat sink or a residual heat removal loop. This redundancy ensures that a single failure will not result in a complete loss of decay heat removal. Above 350°F, the normal RHR suction line is isolated from the RCS to protect RHR piping from overpressurization due to inadvertent SI pump actuation.

During operating modes when the reactor coolant $T_{avg}$ is less than 200°F, but not in the refueling operation condition, a sufficient decay heat removal capability is provided by a residual heat removal loop.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) containment spray with sodium hydroxide addition. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 95°F).[4] In the event of a Design Basis Accident, any one of the following configurations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting off-site doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also in the event of a Design Basis Accident, any one of three configurations of fan-cooler units (with charcoal filters) and/or containment spray pumps (with sodium hydroxide addition) will reduce airborne organic and molecular iodine activities sufficiently to limit off-site doses to acceptable values.[5] Any one of these three configurations constitutes the minimum safeguards for iodine removal.

The combination of three fan-coolers and one containment spray pump is capable of being operated on emergency power with one diesel generator failing to start. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of off-site power or operation of all emergency diesel generators.

A.1

4.5 TESTS FOR ENGINEERED SAFETY FEATURES AND AIR FILTRATION SYSTEMS (A.2)

~~Applicability~~

~~Applies to testing of the Safety Injection System, the Containment Spray System, the Hydrogen Recombiner System, and the Air Filtration Systems.~~

~~Objective~~

~~To verify that the subject systems will respond promptly and perform their design functions, if required.~~

Specification

A. SYSTEM TESTS

1. Safety Injection System

SR 3.5.2.4
SR 3.5.2.5

a. System tests shall be performed at least once per 24 months*. ~~With the Reactor Coolant System pressure less than or equal to 350 psig and temperature less than or equal to 350°F,~~ (LA.3) a test (or actual) (A.3) safety injection signal will be applied to initiate operation of the system. ~~The safety injection and residual heat removal pumps are made inoperable for this test.~~ (A.5)

SR 3.5.2.4
SR 3.5.2.5

b. The test will be considered satisfactory if ~~control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing, that is,~~ (LA.3) the appropriate pump ~~breakers shall have opened and closed~~ (starts) and the appropriate valves shall have completed their travel.

~~c.~~ ~~Conduct a flow test of the high head safety injection system after any modification is made to either its piping and/or valve arrangement.~~ (A.6)

SR 3.5.2.6

~~d.~~ Verify that the mechanical stops on Valves 856 A, C, D, E, F, H, J and K are set ~~at the position measured and recorded during the most recent ECCS operational flow test or flow tests performed in accordance with (c) above.~~ (LA.3) ~~This surveillance procedure shall be performed following any maintenance on these valves or their associated motor operators~~ (A.6) and at a convenient outage if the position of the mechanical stops have not been verified in the preceding three months. (L.5)

SEE ITS 3.8.1 * The time delay relays will be tested at intervals no greater than 22.5 months (18 months + 25%).





Add SR 3.5.2.7 (M.2)

SEE ITS 3.6.6

SEE ITS 3.7.8

LA2

In accordance with Inservice Test Program

B. Component Tests

1. Pumps

SR 3.5.2.3

a. The safety injection pumps, residual heat removal pumps, containment spray pumps and the auxiliary component cooling water pumps shall be started at quarterly intervals. The recirculation pumps shall be started ~~at least once per 24 months~~

b. Acceptable levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, ~~and operate for at least fifteen minutes~~

2. Valves

SEE ITS 3.6.7

a. Each spray additive valve shall be cycled by operator action with the pumps shut down at least once per 24 months.

SEE ITS 3.5.1

b. The accumulator check valves shall be checked for operability at least once per 24 months.

SEE ITS 3.4.14

c. The following check valves shall be checked for gross leakage at least once per 24 months:

| | | | |
|---|---|---|---|
| 857A & G | 857J | 857S & T | 897B |
| 857B | 857K | 857U & W | 897C |
| 857C | 857L | 895A | 897D |
| 857D | 857M | 895B | 838A |
| 857E | 857N | 895C | 838B |
| 857F | 857P | 895D | 838C |
| 857H | 857Q & R | 897A | 838D |

d. In addition to 4.5.B.2.c, the following check valves shall be checked for gross leakage every time the plant is shut down and the reactor coolant system has been depressurized to 700 psig or less. This gross leakage test shall also be performed following valve maintenance, repair or other work which could unseat these check valves:

SEE ITS 3.4.14

| | | |
|---|---|---|
| 838A | 895A | 897A |
| 838B | 895B | 897B |
| 838C | 895C | 897C |
| 838D | 895D | 897D |

Basis

A.1

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally on standby during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes reactor trip, main feedwater isolation and containment isolation, and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is, therefore, to combine systems tests to be performed during plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action and verification made that the components receive the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.(1)

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly (in accordance with Specification 4.1). The testing of the analog channel inputs is accomplished in the same manner as for the reactor protection system. The engineered safety features logic system is tested by means of test switches to simulate inputs from the analog channels. The test switches allow actuation of the master relay, while at the same time blocking the slave relays. Verification that the logic is accomplished is indicated by the matrix test light. The slave relay coil circuits are continuously verified by a built-in monitoring circuit. In addition, the active components (pumps and valves) are to be tested in accordance with the Indian Point 3 Inservice Testing Program. The pumps, specified in the Technical Specifications, are tested on a quarterly basis to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The exception to this quarterly test are the recirculation pumps which are tested during a refueling outage. The quarterly test interval is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent testing would result in increased wear over a long period of time.

A.1

~~Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System, and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification 4.1, the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation, and by these means are continuously monitored for satisfactory performance.~~

~~The charcoal portion of the containment air recirculation system is a passive safeguard which is isolated from the cooling air flow during normal reactor operation. Hence, the charcoal should have a long useful lifetime. The filter frames that house the charcoal are stainless steel and should also last indefinitely. However, the visual inspection specified in Section A.4(a) of this specification will be performed to verify that this is, in fact, the case. The iodine removal efficiency cannot be measured with the filter cells in place. Therefore, at periodic intervals a representative sample of charcoal is to be removed and tested to verify that the efficiencies for removal of methyl iodide are obtained.[2] The fuel storage building air treatment system is designed to filter the discharge of the fuel storage building atmosphere to the facility vent during normal conditions. As required by Specifications 3.8.A.12 and 3.8.C.6, the fuel storage building emergency ventilation system must be operable whenever irradiated fuel is being moved. However, if the irradiated fuel has had a continuous 45-day decay period, the fuel storage building emergency ventilation system is not technically necessary, even though the system is required to be operable during all fuel handling operations. The emergency ventilation fan is automatically started upon high radiation signal and since the bypass assembly is sealed by manually operated isolation devices, air flow is directed through the emergency ventilation HEPA filters and charcoal adsorbers.~~

~~High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of these adsorbers for all emergency air treatment systems. The charcoal adsorbers are installed to reduce the potential release of radio-iodine to the environment. The in-place test results should indicate a system leak tightness of less than or equal to one percent leakage for the charcoal adsorbers and a HEPA efficiency of greater than or equal to 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of greater than or equal to 90 percent on the fuel handling system samples, and greater than or equal to 85 percent on the containment system samples for expected accident conditions. With the efficiencies of the HEPA filters and charcoal adsorbers as specified, further assurance is provided that the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed.~~

~~The basis for the toxic gas monitoring system is given in Technical Specification Section 3.3.~~

~~The control room air treatment system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room air treatment system is designed to automatically start upon control room isolation.~~

A.1

~~For the eight flow distribution valves (856 A, C, D, E, F, H, J and K), verification of the valve mechanical stop adjustments is performed periodically to provide assurance that the high head safety injection flow distribution is in accordance with flow values assumed in the core cooling analysis.~~

~~Gross leakage testing of the reactor coolant system pressure isolation valves and the Low Pressure Injection(LPI)/residual heat removal(RHR)system valves reduces the probability of an inter-system LOCA[4]. These tests implement the requirements set forth in NRC generic letter dated February 23, 1980, regarding testing of LPI/RHR system check valves. This amendment provides a basis for the rescission of item A.5. of a Confirmatory Order issued by the Commission to Indian Point 3 in a letter dated, February 11, 1980. To satisfy ALARA requirements, gross leakage (>10 gpm) may be measured indirectly (i.e. using installed pressure and flow indications).~~

~~References~~


~~(1) FSAR Section 6.2~~

~~(2) FSAR Section 6.4~~

~~(3) FSAR Section 6.8~~

~~(4) WASH 1400~~

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.5.2:
# "ECCS - Operating"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.5.2 - ECCS - Operating

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 4.5.A.1.a states that a test safety injection signal will be applied to initiate operation of the safety injection system for the system test performed every 24 months. ITS SR 3.5.2.4 and ITS SR 3.5.2.5 allow the use of either an actual or simulated actuation signal to verify valve actuation and pump start on receipt of a safety injection actuation signal. This change is acceptable because use of an actual instead of a simulated or "test" signal will not affect the performance of the test because the equipment being tested cannot discriminate between an actual or simulated signal. This is an administrative change with no impact on safety because the use of an actual or simulated signal does not change the validity of the test as a verification of plant response to the event.

A.4 CTS 3.3.A.4.b and CTS 3.3.A.4.c allow one safety injection pump or one residual heat removal pump, respectively, to be out of service for a specified allowable out of service time (See ITS 3.5.2, DOC L.2). CTS 3.3.A.4.d allows one residual heat exchanger to be out of service for a specified allowable out of service time (See ITS 3.5.2, DOC L.2). CTS 3.3.A.4.e allows any valve required for the functioning of the safety injection and residual heat removal systems during and following accident conditions to be inoperable for a specified allowable out of service time (See ITS 3.5.2, DOC L.2) provided all valves in the system that provide a duplicate function are operable. CTS 3.3 does not provide an allowed outage time for the recirculation pumps (See ITS 3.5.2, DOC L.2).

Under the same conditions, ITS 3.5.2, Required Action A.1, allows one or more trains to be inoperable (e.g., HHSI can be inoperable in one train and RHR can be inoperable in the same or a different train) with a specified allowable out of service time provided at least 100% of the ECCS flow equivalent (for each ECCS system - HHSI, RHR and recirculation) to operable ECCS trains is available. Except as addressed in Discussions of Change L.1 and L.2, the change in the presentation of requirements is an administrative change with no adverse impact on safety.

A.5 CTS 4.5.A.1.a establishes requirements for a functional test of the ECCS subsystems and includes the statement that the "safety injection and residual heat removal pumps are made inoperable for this test." This statement is deleted because ITS includes appropriate controls and guidance for the determination of system Operability during testing. This is an administrative change with no adverse impact on safety.

A.6 CTS 4.5.A.1.c and CTS 4.5.A.1.d include requirements to perform required Surveillances as part of post maintenance testing specified ECCS pumps and valves. This requirement is not included in ITS because ITS SR 3.0.1 requires that SRs are met whenever equipment is required to be Operable. The Bases for SR 3.0.1 include the clarification that upon completion of maintenance, appropriate post maintenance testing is required to declare equipment Operable. This includes ensuring applicable Surveillances are not failed. Therefore, CTS statements establishing requirements to verify SRs are met following maintenance can be deleted. This is an administrative change with no adverse impact on safety.

A.7 CTS 3.3.A.3 establishes Applicability for ECCS systems as whenever reactor coolant system temperature is above 350°F; however, when requirements are not met, CTS 3.3.A.5 requires that the plant proceed to cold shutdown.

ITS 3.5.2 maintains the requirement to have ECCS systems Operable above 350°F (Modes 1, 2 and 3); however, when requirements are not met, ITS 3.5.2, Required Action B.2, requires that the plant be placed in Mode 4 (less than 350°F).

This change is needed to establish consistency between the LCO Applicability and the associated Required Actions. This is an administrative change with no impact on safety because the CTS 3.3.A.5 requirement to proceed to cold shutdown would not be applicable after temperature is reduced below 350°F. Therefore, this change has no adverse impact on safety.

MORE RESTRICTIVE

M.1 CTS 3.3 and CTS 4.5.A do not establish any requirements for the periodic verification that each valve in the ECCS flow path is in the correct position even for those valves identified in CTS 3.3.A.3 as required to be de-energized in a specific status.

ITS SR 3.5.2.1 is added to require verification every 12 hours that those valves in the ECCS flow path that, if mispositioned, would render more than one train of ECCS inoperable are in the correct position with power removed. A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

ITS SR 3.5.2.2 is added to require verification every 31 days of correct alignment of any manual, power operated, and automatic valves in the ECCS flow paths that are not locked, sealed, or otherwise secured in position. The 31 day Frequency is appropriate because the valves are operated under administrative controls and an improper valve position affects only a single train. This Frequency has been shown to be acceptable through operating experience.

These changes are acceptable because they do not introduce any operation that is un-analyzed while requiring periodic verification that analysis assumptions regarding the Operability of ECCS system flow paths are satisfied. Therefore, this change has no significant adverse impact on safety.

M.2 CTS 3.3 and CTS 4.5.A do not establish any requirements for the periodic verification that containment sump suction inlets are unrestricted and otherwise in proper operating condition.

ITS SR 3.5.2.7 is added to require verification every 24 months that containment sump and recirculation sump suction inlets are unrestricted and otherwise in proper operating condition. This Frequency is consistent with the need to perform this verification while the plant is shutdown and, based on industry experience, is sufficient to detect abnormal degradation.

This change is acceptable because it does not introduce any operation that is un-analyzed while requiring periodic verification that analysis assumptions regarding the Operability of ECCS system flow paths are satisfied. Therefore, this change has no significant adverse impact on safety.

M.3 CTS 3.3.A.5 establishes the Actions required if the ECCS systems (Refueling Water Storage Tank, Accumulators, Safety Injection Pumps, Residual Heat Removal Pumps, Recirculation Pumps) are not restored to meet CTS requirements within specified completion times when above 350°F (Mode 3). CTS 3.3.A.5.a specifies that, if the reactor is critical when requirements are not met, then the reactor must be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within the following 24 hours.

Under the same conditions, ITS 3.5.2, Required Actions B.1 and B.2, require that the reactor be in Mode 3 in 6 hours (See ITS 3.5.1, DOC L.6) and in Mode 4 within 12 hours. This is a more restrictive change because ITS 3.5.2, Required Action B.2, places the plant outside of the LCO Applicability within 12 hours whereas CTS 3.3.A.5.a could allow the plant to stay within the LCO Applicability for 24 hours.

This change is needed to ensure that the plant is placed outside the LCO Applicability promptly when the LCO requirements are not met. This change is acceptable because placing the plant outside the LCO Applicability when LCO requirements are not met is conservative. Additionally, 12 hours is a reasonable time to reach the required plant conditions (Mode 4) from full power conditions in an orderly manner and without challenging plant systems. Therefore, this change has no significant adverse impact on safety.

M.4 CTS 3.3.A.5 establishes the Actions required if the ECCS systems (Refueling Water Storage Tank, Accumulators, Safety Injection Pumps, Residual Heat Removal Pumps, Recirculation Pumps) are not restored to meet CTS requirements within specified completion times when above 350°F (Mode 3). CTS 3.3.A.5.a specifies that, if the reactor is critical when requirements are not met, then the reactor must be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within the following 24 hours. However, if the reactor is subcritical when requirements are not met, CTS 3.3.A.5.b requires only that reactor coolant system temperature and pressure not be increased more than 25°F and 100 psi, respectively, over existing values with the requirement to proceed to cold shutdown (Mode 5) deferred by 48 hours.

Under the same conditions, ITS 3.5.2, Required Actions B.1 and B.2, require that the reactor be in Mode 3 in 6 hours (See ITS 3.5.1, DOC M.3) and in Mode 4 within 12 hours regardless of the status of the unit when the Condition is identified. The allowance provided in CTS 3.3.A.5.b is deleted.

This change is needed to eliminate the ambiguity created by CTS 3.3.A.5.b when performing a reactor shutdown and cooldown required by CTS 3.3.A.5 and to ensure that the plant is placed outside the LCO Applicability promptly when the LCO requirements are not met. This change is acceptable because placing the plant outside the LCO Applicability when LCO requirements are not met is conservative and there is no change in the CTS 3.3.A.5 requirement. This change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.3.A.4 limits the number of concurrent inoperable ECCS systems (Refueling Water Storage Tank, Accumulators, High Head Safety Injection Pumps (HHSI), Residual Heat Removal (RHR) Pumps, Recirculation Pumps) by allowing "any one" of these ECCS systems to be inoperable "at any one time." Therefore, in addition to the specific directions provided in CTS 3.3.A.4.a through CTS 3.3.A.4.g, CTS 3.3.A.4 does not permit concurrent inoperability of the RWST, Accumulators, HHSI Pumps, RHR Pumps, or Recirculation Pumps.

ITS LCO 3.5.1, ECCS Accumulators, ITS LCO 3.5.2, ECCS Systems - Operating, and ITS LCO 3.5.4, RWST, do not establish any restrictions on the concurrent inoperability of the RWST, Accumulators, High Head Safety Injection Pumps, Residual Heat Removal Pumps, and Recirculation Pumps.

This change is acceptable because 2 of the 3 HHSI pumps, 1 of the 2 RHR pumps, and 1 of the 2 recirculation pumps is the minimum complement of ECCS systems assumed available is the safety analysis and this minimum complement is sufficient to mitigate a design basis event at IP3. Additionally, each of these ECCS systems provides a different safety function; therefore, more than the minimum required number of pumps for any of these systems does not provide significant compensation for an inoperable pump in an ECCS system that provides a different safety function. Finally, the very short (one hour) allowable out of service time (AOT) when an ECCS accumulator or the RWST is inoperable due to a critical feature not within limits (See ITS 3.5.4, DOC L.2 and ITS 3.5.1, DOC L.2) is not affected by a concurrent inoperability of another ECCS system in either the CTS or the ITS. Therefore, elimination of the restriction in CTS 3.3.A.4 that prohibits concurrent inoperable ECCS systems has no significant adverse consequence and is deleted.

L.2 CTS 3.3.A.4.b and CTS 3.3.A.4.c allow one safety injection pump or one residual heat removal pump, respectively, to be inoperable with an allowable out of service time (AOT) of 24 hours. CTS 3.3.A.4.d allows one RHR heat exchanger to be inoperable with an AOT of 48 hours. CTS 3.3.A.4.e allows any valve required for the functioning of the safety injection and residual heat removal systems to be inoperable with an AOT of 24 hours provided all valves in the system that provide a duplicate function are Operable. CTS 3.3 does not provide an allowed outage time for the recirculation pumps.

Under the same conditions, ITS 3.5.2, Required Action A.1, allows one or more trains to be inoperable (e.g., HHSI can be inoperable in one train and RHR can be inoperable in the same or a different train, etc.) with an allowable out of service time of 72 hours provided at least 100% (for each ECCS system - HHSI, RHR and recirculation) of the ECCS flow equivalent to operable ECCS trains is available. This change creates a new 72 hour AOT for recirculation subsystems and extends the AOT for other ECCS subsystems to 72 hours. The 72 hour AOT is applicable only if a combination of equipment remains Operable such that 100% (for each ECCS system - HHSI, RHR and recirculation) of the ECCS flow equivalent to operable ECCS trains is available. This change is supported by ITS LCO 3.8.1, Required Actions, which limits the time that an ECCS component may be inoperable if the normal or emergency power supply to the redundant component is inoperable.

This change is acceptable based on a reliability analysis discussed in NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975, which concluded that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours. Additionally, with one or more component(s) inoperable such that 100% of the flow equivalent to OPERABLE ECCS trains is not available, the facility is in a condition outside the accident analysis and LCO 3.0.3 is entered. This change has no significant adverse impact on safety.

L.3 ITS LCO 3.5.2, Note 1, provides a new allowance that both ECCS injection flow paths may be isolated when in Mode 3 by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1. CTS includes no such allowance.

This change is needed because ITS SR 3.4.14.1 includes a new requirement that pressure isolation valve testing per SR 3.4.14.1 must be performed within 24 hours following valve actuation due to automatic or manual action or if there is flow through the valve. This allows performance of required testing in Mode 3 and facilitates timely completion for return to power Operation.

This change is acceptable because of the stable conditions associated with operation in Mode 3, the low probability of occurrence of a Design Basis Accident (DBA) during the period the flow paths are isolated, the limited core cooling requirements in Mode 3, and because the required flow paths are either readily restorable from the control room or the valves are closed under administrative controls that ensure prompt restoration if required. Therefore, this change has no significant adverse impact on safety.

L.4 ITS LCO 3.5.2, Note 2, provides a new allowance that operation in Mode 3 with ECCS pumps declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is allowed for up to 4 hours or until the temperature of all RCS cold legs exceeds 375°F, whichever comes first. CTS includes no such allowance.

This change is needed because the IP3 LTOP enable temperature (currently 319°F) is close enough to the Mode 3 boundary temperature of 350°F that only a small window exists for the restoration from LTOP requirements during heatup. This allowance allows temperature to be established safely above the LTOP enable temperature before restoration is performed and verified without impacting plant heatup.

This change is acceptable because the Mode 3 condition when this allowance can be used is restricted to less than 375°F, the duration in Mode 3 without the full complement of ECCS subsystems is limited to 4 hours, the low probability of occurrence of a Design Basis Accident (DBA) during the 4 hour period, and the limited core cooling requirements in Mode 3 during a plant startup. Therefore, this change has no significant adverse impact on safety.

L.5 CTS 4.5.A.1.d requires verification that the stops on the high head safety injection valves are correctly set at a convenient outage if the position of the stops have not been verified in the preceding three months (otherwise the 24 month SR Frequency in CTS 4.5.A.1.a is applicable).

ITS SR 3.5.2.6 requires verification that the stops on the high head safety injection valves are correctly set at a 24 month Frequency (See ITS 3.5.2, DOC A.6 for post maintenance requirements).

Elimination of the special SR Frequency for verification that the stops on the high head safety injection valves are correctly set is acceptable based on the demonstrated reliability of these valves and the stops. Operating experience has confirmed that a 24 month SR Frequency provides a high degree of assurance that the stops will be maintained in the required position. Therefore, this change has no significant adverse impact on safety.

L.6 CTS 3.3.A.5 establishes Actions required if ECCS systems (Refueling Water Storage Tank, Accumulators, Safety Injection Pumps, Residual Heat Removal Pumps, Recirculation Pumps) are not restored to meet CTS requirements within specified completion times when above 350°F (Mode 3). CTS 3.3.A.5.a specifies that, if the reactor is critical when requirements are not met, then the reactor shall be in hot shutdown (Mode 3) within 4 hours.

Under the same conditions, ITS 3.5.2, Required Action B.1, requires that the reactor be in Mode 3 in 6 hours. ITS 3.5.2, Required Action B.1, extends the time allowed to reach Mode 3 when requirements are not met from 4 hours to 6 hours.

This change is needed because 6 hours is a reasonable time, based on operating experience, to reach the required plant conditions (Mode 3) from full power conditions in an orderly manner and without challenging plant systems. This change is acceptable because of the low probability of a DBA occurring during the additional 2 hours allowed to reach Mode 3. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

LA.1 CTS 3.3.A.3.e, f, and g require Operability of three safety injection pumps, two residual heat removal pumps and heat exchangers, and two recirculation pumps with the associated piping and valves whenever the reactor is above 350°F (Modes 1, 2 and 3).

ITS 3.5.2 requires Operability of three ECCS trains with the ECCS trains defined in the ITS 3.5.2 Bases and system descriptions in the FSAR. Specifically, the ITS 3.5.2 Bases specify that the ECCS Function is provided by three separate ECCS systems: high head safety injection (HHSI), residual heat removal (RHR) injection, and containment recirculation. Each ECCS system is divided into subsystems as follows: three 50% capacity HHSI subsystems; two 100% capacity RHR subsystems; and, two 100% capacity recirculation subsystems. Each of these subsystems is further described as including the required valves, heat exchangers and flow paths needed for the subsystem to perform its safety function. Furthermore, the ECCS subsystems (3 HHSI, 2 RHR and 2 Recirculation) are grouped into three trains (5A, 2A/3A and 6A) such that any two of the 3 trains are capable of meeting all ECCS capability assumed in the accident analysis. The ITS 3.5.2 ECCS trains use the same designation as the Safeguards Power Trains required by LCO 3.8.9, Distribution Systems - Operating, with Safeguards Power Train 5A supported by DG 33, Safeguards Power Train 2A/23 supported by DG 31, Safeguards Power Train 6A supported by DG 32.

Establishing ECCS requirements in terms of ECCS trains with the ECCS subsystems and trains defined in the ITS 3.5.2 Bases is needed because this presentation ensures that requirements are clearly understood and consistently applied in conjunction with ITS 3.8.1, AC Sources - Operating, and ITS 3.8.9, Distribution Systems - Operating.

This change is acceptable because ITS 3.5.2 maintains the existing requirement for the Operability of three trains of ECCS; therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

This change, which allows the description of the design of the ECCS systems to be maintained in the FSAR and the detailed description of the requirements for Operability of these systems to be maintained in the ITS Bases, is consistent with the approach used in NUREG-1431 for all Limiting Conditions for Operation (LCOs). This approach is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, and ITS 5.5.13, Technical Specifications (TS) Bases Control Program, are designed to assure that changes to the FSAR and ITS Bases do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 and ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of FSAR and Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

LA.2 CTS 4.5.B.1 requires starting the pump and operating for at least 15 minutes at the required pressure every quarter for the safety injection pumps, residual heat removal pumps, containment spray pumps and the auxiliary component cooling water pumps and every 24 months for the recirculation pumps.

ITS SR 3.5.2.3 maintain the requirements to verify each ECCS pump's developed head is greater than or equal to the required head; however, the Frequency is specified as in accordance with the Inservice Testing (IST) Program. Additionally, the requirement to run each pump for 15 minutes is also relocated to the IST. The Inservice Test (IST) Program is required by ITS 5.5.7 and provides controls for inservice testing of ASME Code Class 1, 2, and 3 components.

ITS 5.5.7, Inservice Testing Program (IST), requires establishing and maintaining a program for inservice testing of ASME Code Class 1, 2, and 3 components at frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code. Additionally, 10 CFR 50.55a(f) already provides the regulatory requirements for this IST Program, and specifies that ASME Code Class 1, 2, and 3 pumps and valves are covered by an IST Program. Therefore, maintaining the requirement that ECCS pumps must be Operable in ITS 3.5.2 and maintaining the requirement for periodic testing of pumps in the IST Program required by ITS 5.5.7 provides a high degree of assurance that ECCS systems will be tested and maintained to ensure ECCS Operability. Additionally, ITS 5.5.7, Inservice Testing Program (IST), requirements and 10 CFR 50.55a(f) ensure adequate change control and regulatory oversight for any changes to the existing requirements. Therefore, requirements to test ECCS pumps can be maintained in the ITS with the Frequency in the IST program with no significant adverse impact on safety.

LA.3 CTS 4.5.A.1.a, CTS 4.5.A.1.b, and CTS 4.5.A.1.d include detail regarding ECCS test conditions, test performance and test acceptance criteria.

ITS SR 3.5.2.4, ITS SR 3.5.2.5 and ITS SR 3.5.2.4 maintain the requirements for testing ECCS subsystem initiation and positioning of stops for HHSI injection valves; however, detail regarding safety injection system test conditions, test performance and test acceptance criteria are moved to the ITS 3.5.2 Bases.

This is acceptable because the requirements to perform the safety injection system tests are maintained in ITS SR 3.5.2.4, ITS SR 3.5.2.5, and ITS SR 3.5.2.6. The particular details about test conditions, test performance and test acceptance criteria being moved to the Bases are not essential elements for performing a test that verifies ECCS Operability. Furthermore, the ITS 5.5.13, Technical Specifications (TS) Bases Control Program, is designed to assure that changes to the ITS Bases do not result in changes to the Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

LA.4 CTS 3.3.A.3.n requires that RCS temperature must not exceed 350°F unless the RHR system is in the ESF alignment with the normal RHR suction line isolated from the RCS. This requirement is intended to protect the RHR system from over pressurization. This valve lineup is not included in the ITS 3.5.2 and is maintained in the FSAR and implemented by procedures. This change is acceptable because this requirement is backed up by the RHR auto-closure interlock and ITS SR 3.4.14.2 and SR 3.4.14.3 maintain the requirement to verify the automatic isolation and interlock function for RHR isolation valves every 24 months. This function prevents the RHR isolation valves from being opened with an RCS pressure signal ≥ 450 psig, and causes the valves to close automatically with RCS pressure signal ≥ 550 psig.

Maintaining the valve lineup requirement in the FSAR (and implemented by procedures) is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, are designed to assure that changes to the FSAR do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.2:
## "ECCS - Operating"

## PART 4:

## No Significant Hazards Considerations
## for
## Changes between CTS and ITS
## that are
## Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.5.2 - ECCS - Operating

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS LCO 3.5.1, ECCS Accumulators, ITS LCO 3.5.2, ECCS Systems-Operating, and ITS LCO 3.5.4, RWST, eliminates the CTS 3.3.A.4 restrictions on the concurrent inoperability of the RWST, Accumulators, High Head Safety Injection Pumps, Residual Heat Removal Pumps, and Recirculation Pumps.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the Operability status of ECCS systems has no affect on the initiators of any analyzed events.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because all 4 accumulators, 2 of the 3 HHSI pumps, 1 of the 2 RHR pumps, and 1 of the 2 recirculation pumps is the minimum complement of ECCS systems assumed available in the safety analysis and this minimum complement is sufficient to mitigate a design basis. Additionally, each of these ECCS systems provides a different safety function; therefore, more than the minimum required number of pumps and/or accumulators for any of these systems does not provide significant compensation for an inoperable pump and/or accumulator in an ECCS system that provides a different safety function. Finally, the very short (one hour) allowable out of service time (AOT) when an ECCS accumulator or the RWST is inoperable due to a critical feature not within limits is not affected by a concurrent inoperability of another ECCS system in either the CTS or the ITS.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS systems are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because all 4 accumulators, 2 of the 3 HHSI pumps, 1 of the 2 RHR pumps, and 1 of the 2 recirculation pumps is the minimum complement of ECCS systems assumed available in the safety analysis and this minimum complement is sufficient to mitigate a design basis. Additionally, each of these ECCS systems provides a different safety function; therefore, more than the minimum required number of pumps and/or accumulators for any of these systems does not provide significant compensation for an inoperable pump and/or accumulator in an ECCS system that provides a different safety function. Finally, the very short (one hour) allowable out of service time (AOT) when an ECCS accumulator or the RWST is inoperable due to a critical feature not within limits is not affected by a concurrent inoperability of another ECCS system in either the CTS or the ITS.

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LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change creates a new 72 hour AOT for recirculation subsystems and extends the AOT for other ECCS subsystems to 72 hours. The 72 hour AOT is applicable only if a combination of equipment remains Operable such that 100% (for each ECCS system - HHSI, RHR and recirculation) of the ECCS flow equivalent to operable ECCS trains is available.

This change will not result in a significant increase in the probability of an accident previously evaluated because the Operability status of ECCS systems has no affect on the initiators of any analyzed events.

This change will not result in a significant increase in the consequences of an accident previously evaluated because a reliability analysis discussed in NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975, concluded that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours. Additionally, with one or more component(s) inoperable such that 100% of the flow equivalent to OPERABLE ECCS trains is not available, the facility is in a condition outside the accident analysis and LCO 3.0.3 must be immediately entered.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS systems are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because a reliability analysis discussed in NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975, concluded that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours. Additionally, with one or more component(s) inoperable such that 100% of the flow equivalent to OPERABLE ECCS trains is not available, the facility is in a condition outside the accident analysis and LCO 3.0.3 must be immediately entered.

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LESS RESTRICTIVE
("L.3" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS LCO 3.5.2, Note 1, provides a new allowance that both ECCS injection flow paths may be isolated when in Mode 3 by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1. CTS includes no such allowance.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the Operability status of ECCS systems has no affect on the initiators of any analyzed events.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because of the stable conditions associated with operation in Mode 3, the low probability of occurrence of a Design Basis Accident (DBA) during the period the flow paths are isolated, the limited core cooling requirements in Mode 3, and because the required flow paths are either readily restorable from the control room or the valves are closed under administrative controls that ensure prompt restoration if required.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS systems are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because of the stable conditions associated with operation in Mode 3, the low probability of occurrence of a Design Basis Accident (DBA) during the period the flow paths are isolated, the limited core cooling requirements in Mode 3, and because the required flow paths are either readily restorable from the control room or the valves are closed under administrative controls that ensure prompt restoration if required.

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LESS RESTRICTIVE
("L.4" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS LCO 3.5.2, Note 2, provides a new allowance that operation in Mode 3 with ECCS pumps declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is allowed for up to 4 hours or until the temperature of all RCS cold legs exceeds 375°F, whichever comes first. CTS includes no such allowance.

This change will not result in a significant increase in the probability of an accident previously evaluated because the Operability status of ECCS systems has no affect on the initiators of any analyzed events.

This change will not result in a significant increase in the consequences of an accident previously evaluated because the Mode 3 condition when this allowance can be used is restricted to less than 375°F, the duration in Mode 3 without the full complement of ECCS subsystems is limited to 4 hours, the low probability of occurrence of a Design Basis Accident (DBA) during the 4 hour period, and the limited core cooling requirements in Mode 3 during a plant startup.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS systems are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the Mode 3 condition when this allowance can be used is restricted to less than 375°F, the duration in Mode 3 without the full complement of ECCS subsystems is limited to 4 hours, the low probability of occurrence of a Design Basis Accident (DBA) during the 4 hour period, and the limited core cooling requirements in Mode 3 during a plant startup.

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LESS RESTRICTIVE
("L.5" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS SR 3.5.2.7 requires verification that the mechanical stops on the high head safety injection valves are correctly set at a 24 month Frequency and eliminates the requirement that this verification is performed at every convenient outage if the position of the mechanical stops have not been verified in the preceding three months.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the status of the mechanical stops on the high head safety injection valves has no affect on the initiators of any analyzed events.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because both the design reliability of these valves and the mechanical stops and operating experience have confirmed that a 24 month SR Frequency provides a high degree of assurance that mechanical stops will be maintained in the required position.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS systems are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because both the design reliability of these valves and the mechanical stops and operating experience have confirmed that a 24 month SR Frequency provides a high degree of assurance that mechanical stops will be maintained in the required position.

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LESS RESTRICTIVE
("L.6" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS 3.5.2, Required Action B.1, extends the time allowed to reach Mode 3 when requirements for ECCS systems (Safety Injection Pumps, Residual Heat Removal Pumps and Recirculation Pumps) are not met from 4 hours to 6 hours.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the Operability status of ECCS systems has no affect on the initiators of any analyzed events.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because of the low probability of a DBA occurring during the additional 2 hours allowed to reach Mode 3 and because 6 hours is a reasonable time, based on operating experience, to reach the required plant conditions (Mode 3) from full power conditions in an orderly manner and without challenging plant systems.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS systems are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because of the low probability of a DBA occurring during the additional 2 hours allowed to reach Mode 3 and because 6 hours is a reasonable time, based on operating experience, to reach the required plant conditions (Mode 3) from full power conditions in an orderly manner and without challenging plant systems.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.5.2:
# "ECCS - Operating"

# PART 5:

# NUREG-1431
# Annotated to show differences between NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.5.2

This ITS Specification is based on NUREG-1431 Specification No. **3.5.2**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| WOG-063 | **153 R0** | CLARIFY EXCEPTION NOTES TO BE CONSISTENT WITH THE REQUIREMENT | Approved by NRC | Incorporated | **T.1** |
| WOG-067 R1 | | RELOCATE LTOP ARMING TEMPERATURE TO PTLR | Rejected by TSTF | Not Incorporated | **N/A** |

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

<Doc LA> **3.5.2 ECCS—Operating** DB.2

<3.3.A.3.e>
<3.3.A.3.f>
<3.3.A.3.g>

LCO 3.5.2 Three ~~Two~~ ECCS trains shall be OPERABLE. T.1

<3.3.A.3> APPLICABILITY: MODES 1, 2, and 3.

made incapable of injecting

---NOTES---

<DOC L.3>

1. In MODE 3, both HHSI ~~safety injection (SI) pump~~ flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.

<DOC L.4> T.1

2. Operation in MODE 3 with HHSI ~~ECCS~~ pumps ~~declared inoperable~~ made incapable of injecting pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) ~~System~~," is allowed for up to 4 hours or until the temperature of all RCS cold legs exceeds [375]°F, whichever comes first.

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ACTIONS

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <3.3.A4> <DOC L.1> <3.3.A.4.b> <3.3.A.4.c> <3.3.A.4.d> <3.3.A.4.e> <DOC L.2> | A. One or more trains inoperable.<br>**AND**<br>At least 100% of the ECCS flow equivalent to ~~a single~~ OPERABLE ECCS train Δ available. | A.1 Restore train(s) to OPERABLE status. | 72 hours DB.2 |
| <3.3.A5> <DOC M.3> <DOC M.4> <DOC L.6> | B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br>**AND**<br>B.2 Be in MODE 4. | 6 hours<br><br>12 hours |

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| <DOC M.1> <3.3.A.3.h> <3.3.A.3.i> <3.3.A.3.j> <3.3.A.3.l> <3.3.A.3.m> | [SR 3.5.2.1 Verify the following valves are in the listed position with power to the valve operator removed.<br>Insert: 3.5-5-01<br>~~Number / Position / Function~~<br>~~[ ] [ ] [ ]~~<br>~~[ ] [ ] [ ]~~<br>~~. . .~~<br>~~[ ] [ ] [ ]~~] | [12 hours] CLB.1 |
| <DOC M.1> | SR 3.5.2.2 Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. | 31 days |
| | ~~SR 3.5.2.3 Verify ECCS piping is full of water.~~ | ~~31 days~~ |
| <4.5.B.1> <LA.2> | SR 3.5.2.~~4~~ 3 Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head. | In accordance with the Inservice Testing Program |
| <4.5.A.1.a> <4.5.A.1.b> <DOC LA3> | SR 3.5.2.~~5~~ 4 Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. | ~~[18]~~ 24 months |

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.5.2 - ECCS - Operating

<u>INSERT: 3.5-5-01</u>:

| Number | Position | Function |
|---|---|---|
| SI-856B | Closed | HHSI Loop 33 Hot Leg Injection Stop Valve |
| SI-856G | Closed | HHSI Loop 31 Hot Leg Injection Stop Valve |
| SI-1810 | Open | RWST outlet isolation |
| AC-744 | Open | Common discharge isolation for RHR pumps |
| SI-882 | Open | Common RWST suction isolation for RHR pumps |
| SI-842 | Open | HHSI pump minimum flow line isolation |
| SI-843 | Open | HHSI pump minimum flow line isolation |
| SI-883 | Closed | RHR pump return to RWST isolation |
| AC-1870 | Open | RHR pump minimum flow line isolation |
| AC-743 | Open | RHR pump minimum flow line isolation |

SURVEILLANCE REQUIREMENTS (continued)

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| <4.5.A.1.a> <4.5.A.1.b> <DOC LA.3> <DOC A.3> | SR 3.5.2.~~6~~ 5 Verify each ECCS pump starts automatically on an actual or simulated actuation signal. | ~~18~~ 24 months |
| <4.5.A.1.d> <DOC L.5> <DOC LA.3> | [SR 3.5.2.~~7~~ 6 Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. Valve Number~~s~~ [ ] [ ] . . . [ ] Insert: 3.5-6-01 | ~~18~~ 24 months] |
| <DOC M.2> | SR 3.5.2.~~8~~ 7 Verify, by visual inspection, each ECCS train containment sump suction inlet and recirculation sump suction inlet is not restricted by debris and the suction inlet ~~trash racks and~~ screens show no evidence of structural distress or abnormal corrosion. DB.2 | ~~18~~ 24 months |

<u>INSERT: 3.5-6-01</u>:

| | |
|---|---|
| SI-856A | SI-856F |
| SI-856C | SI-856H |
| SI-856D | SI-856J |
| SI-856E | SI-856K |

## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.2 ECCS—Operating

BASES

**BACKGROUND**

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident~~, (including uncontrolled steam release or loss of feedwater)~~; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

Recirculation and

recirculation sump or

between 14 and

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After ~~approximately~~ 24 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

Insert: B3.5-10-01

~~The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains.~~ The ECCS accumulators and the RWST are also part of the

H

~~(continued)~~

DB.2

INSERT: B 3.5-10-01:

The ECCS Function is provided by three separate ECCS systems: high head safety injection (HHSI), residual heat removal (RHR) injection, and containment recirculation. Each ECCS system is divided into subsystems as follows:

- HHSI System is divided into three 50% capacity HHSI subsystems. Each HHSI subsystem consists of one pump as well as associated piping and valves to transfer water from the suction source to the core. HHSI subsystem 32 is OPERABLE when capable of injecting using the flow path associated with either HHSI subsystem 31 or 33. Note that the HHSI pumps have a shutoff head of approximately 1500 psig. Therefore, IP3 is classified as a low head safety injection plant.

- RHR injection System is divided into two 100% capacity subsystems. Each ECCS RHR subsystem consists of one RHR pump and one RHR heat exchanger as well as associated piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either RHR subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE RHR injection subsystem.

- Containment Recirculation is divided into two 100% capacity subsystems. Each subsystem consists of one Containment Recirculation pump and one RHR heat exchanger as well as associated piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either Recirculation subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE Containment Recirculation subsystem.

DB.2

**INSERT: B 3.5-10-01**: (continued)

The three ECCS systems (3 HHSI, 2 RHR and 2 Recirculation) are grouped into three trains (5A, 2A/3A and 6A) such that any 2 of the 3 trains are capable of meeting all ECCS capability assumed in the accident analysis. Each ECCS train consists of the following:

a. ECCS Train 5A includes subsystems HHSI 31 and containment recirculation 31;

b. ECCS Train 2A/3A includes subsystems HHSI 32 and RHR 31; and,

c. ECCS Train 6A includes subsystems HHSI 33, RHR 32, and containment recirculation 32.

The ECCS trains use the same designation as the Safeguards Power Trains required by LCO 3.8.9, Distribution Systems - Operating, with Safeguards Power Train 5A supported by DG 33, Safeguards Power Train 2A/23 supported by DG 31, Safeguards Power Train 6A supported by DG 32.

DB.1
DB.2

BASES

BACKGROUND (continued)

ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the high head safety injection ~~centrifugal charging~~ pumps, the RHR pumps, heat exchangers, and the (HH) SI pumps. ~~Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences.~~ Containment recirculation This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from different ~~opposite~~ trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the HHSI and RHR ~~ECCS~~ pumps. ~~Separate piping supplies each subsystem and each train within the subsystem. The discharge from the centrifugal charging pumps combines prior to entering the boron injection tank (BIT) (if the plant utilizes a BIT) and then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg.~~ The discharge from the HHSI and RHR pumps ~~divides and~~ feeds ~~an~~ injection lines to each of the RCS cold legs. Control valves are set to balance the flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

~~For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the centrifugal charging pumps supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.~~ DB.1

Insert: B3.5-11-01

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then ~~supply the other ECCS pumps~~. Initially, recirculation is through the same paths as the injection phase. Subsequently, recirculation alternates injection between the hot and cold legs.

is split

Insert: B3.5-11-02

(continued)

INSERT: B 3.5-11-01: DB.1

the containment recirculation pumps take suction from the containment recirculation sump and direct flow through the RHR heat exchangers to the cold legs. The RHR pumps can be used to provide a backup method of recirculation in which case the

INSERT: B 3.5-11-02: DB.1

supply recirculation flow directly or supply the suction of the HHSI pumps.

BASES

BACKGROUND (continued)

The ~~centrifugal charging subsystem of the~~ ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS HHSI pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

Insert: B3.5-12-01

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.5.2 - ECCS - Operating

INSERT: B 3.5-12-01:

DB.1

,except for the containment recirculation subsystems,

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The ~~centrifugal charging pumps and SI~~ HHSI pumps are credited in a small break LOCA event. ~~This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps.~~ The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one RHR pump (both EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the ~~centrifugal charging and SI pumps~~ HHSI will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. ~~For smaller LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory.~~ For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS ~~trains~~ satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36.

LCO

In MODES 1, 2, and 3, ~~two independent (and redundant)~~ three ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting ~~either~~ any one train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

Insert: B 3.5-14-01

In MODES 1, 2, and 3, ~~an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and automatically transferring suction to the containment sump.~~

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ~~ECCS~~ HHSI and RHR pumps and their respective supply headers to each of the four cold leg injection nozzles. Insert: B 3.5-14-03 In the long term, this flow path may be switched to take its supply from the ~~containment sump and~~ to supply its flow to the RCS hot and cold legs. Insert: B 3.5-14-02

The flow path for each train must maintain its designed independence to ensure that no single failure can disable ~~both~~ more than one ECCS train~~s~~ (except as described in Reference 5). Insert: B3.5-14-04

Insert: From Page 3.5-15

T.1

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.5.2 - ECCS - Operating

<u>INSERT: B 3.5-14-01</u>:

the ECCS consists of the following:

a. ECCS Train 5A includes HHSI subsystem 31 and containment recirculation subsystem 31;

b. ECCS Train 2A/3A includes HHSI subsystem 32 and RHR subsystem 31; and,

c. ECCS Train 6A includes HHSI subsystem 33, RHR subsystem 32, and containment recirculation subsystem 32.

Each HHSI subsystem consists of one pump as well as associated piping and valves to transfer water from the suction source to the core. HHSI subsystem 32 is OPERABLE when capable of injecting using the flow path associated with either HHSI subsystem 31 or 33.

Each ECCS RHR subsystem consists of one RHR pump and one RHR heat exchanger as well as associated piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either RHR subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE RHR injection subsystem.

Each containment recirculation subsystem consists of one Containment Recirculation pump and one RHR heat exchanger as well as associated instrumentation, piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either Recirculation subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE Containment Recirculation subsystem. Note that RHR pump OPERABILITY requires the functional availability of the associated auxiliary component cooling water pump.

DB.1

**INSERT: B 3.5-14-02:**

(8 cold leg injection nozzles for the HHSI pumps)

**INSERT: B 3.5-14-03:**

containment recirculation sump using the containment recirculation pumps or, alternately, the containment sump using the RHR pumps

**INSERT: B 3.5-14-04:**

, either directly into the RCS or via the HHSI pumps.

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements ~~in the lower MODES~~ when at lower power. ~~The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power.~~ The SI HHSI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

This LCO is only applicable in MODE 3 and above. Below MODE 3, ~~the SI signal setpoint is manually bypassed by operator control, and~~ system functional requirements are relaxed as described in LCO 3.5.3, "ECCS—Shutdown."

T.1

As indicated in Note 1, the SI flow path may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. This is acceptable because the flow paths ~~is readily~~ are readily restorable from the control room. (Δ)

Insert: B3.5-15-01

As indicated in Note 2, operation in MODE 3 with ECCS trains ~~declared inoperable~~ made incapable of injecting pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be ~~rendered inoperable~~ at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.

Move to Page B3.5-14

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.~~5~~, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.~~6~~, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level." (4) (5)

(continued)

INSERT: B 3.5-15-01:

or the valves are opened under administrative controls that ensure prompt closure when required. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room.

BASES (continued)

ACTIONS

A.1

(100% HHSI flow, 100% RHR injection flow, and 100% containment recirculation flow)

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to ~~a single~~ OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components. (4)

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

pump

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one ~~component~~ in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different ~~components~~, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to ~~a single~~ two OPERABLE ECCS train remains available. This allows increased flexibility in plant operations under circumstances when ~~components~~ in ~~opposite~~ redundant trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours. (4)

(5)

Reference 6 describes situations in which one more than one component, such as ~~an RHR crossover valve~~ the valves governed by SR 3.5.2.1 can disable ~~both~~ ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent ~~to a single OPERABLE ECCS train~~ for HHSI, RHR and Containment Recirculation is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

more than one

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power or by key locking the control in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference ~~6~~ 5, that can disable the function of both ECCS trains and invalidate the accident analyses. A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.2 (continued)

under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

Insert: B3.5-18-01

~~SR 3.5.2.3~~

~~With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.~~

DB high RWST

SR 3.5.2.~~4~~ 3

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.~~5~~ 4 and SR 3.5.2.~~6~~ 5

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.5.2 - ECCS - Operating

CLB.1

INSERT: B 3.5-18-01:

An exception is the RWST outlet isolation valve. Although closing this valve would render more than one train inoperable, a 31 day Frequency is appropriate because it is a locked manual valve that is located in locked area.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.~~5~~ 4 and SR 3.5.2.~~6~~ 5 (continued)

Insert: B3.5-19-01

simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

24

SR 3.5.2.~~7~~ 6

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves have stops to allow proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. ~~This Surveillance is not required for plants with flow limiting orifices.~~ The 18 month Frequency is based on ~~the same~~ reasons ~~as those~~ stated in SR 3.5.2.~~5~~ 4 and SR 3.5.2.~~6~~ 5.

Insert: B 3.5-19-02

24

SR 3.5.2.~~8~~ 7

Periodic inspections of ~~the~~ each containment and recirculation sump suction inlet ensure that ~~it~~ each is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency ~~has been found to be~~ is sufficient to detect abnormal degradation and is confirmed by industry operating experience.

24

(continued)

**INSERT: B 3.5-19-01:**

Note that the Containment Recirculation system is a manually initiated system and is not included as part of this SR. Additionally,

**INSERT: B 3.5-19-02:**

Therefore, an improperly positioned valve could result in the inoperability of more than one injection flow path. The stops are set based on the results of the most recent ECCS operational flow test.

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.

2. 10 CFR 50.46.

3. FSAR, Section ~~[ ]~~. 14

~~4.~~ ~~FSAR, Chapter [15], "Accident Analysis."~~

4 ~~5.~~ NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

5 ~~6.~~ IE Information Notice No. 87-01.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.2:
## "ECCS - Operating"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.5.2 - ECCS - Operating

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 NUREG 1431, Rev. 1, SR 3.5.2.1 requires verification of proper alignment every 12 hours of any valve that would render more than one ECCS train inoperable if mispositioned. The proper alignment of other valves in the ECCS flowpath is verified every 31 days. IP3 ITS SR 3.5.2.1 and SR 3.5.2.2 differ from NUREG 1431, Rev 1, because the RWST outlet isolation valve, SI846, is verified in its proper position every 31 days. Although closing this valve would render more than one ECCS train inoperable, a 31 day Frequency is appropriate because it is a locked manual valve that is located in locked area. This difference is consistent with the current licensing basis because CTS 3.3 and CTS 4.5.A do not establish any requirements for the periodic verification that each valve in the ECCS flow path is in the correct position even for those valves identified in CTS 3.3.A.3 as required to be de-energized in a specific status. Therefore, IP3 ITS SR 3.5.2.1 and SR 3.5.2.2 require more Frequent verification of the status of the RWST outlet valve than is currently required.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical errors or made a minor editorial improvements to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, these changes are not significant or generic deviations from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.5.2 - ECCS - Operating

DB.2 IP3 ITS 3.5.2 LCO, Conditions and Required Actions, and Bases differ from NUREG 1431, Rev 1, because IP3 uses 3 train of ECCS versus a 2 ECCS train design modeled in the NUREG. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS 3.5.2 Bases. This change maintains the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-153, Rev.0 (WOG-63) which clarifies exception notes to be consistent with the requirement. This is an approved generic change traveler for NUREG-1431.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.5.3:
# "ECCS - Shutdown"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.3 ECCS – Shutdown

LCO 3.5.3 One ECCS residual heat removal (RHR) subsystem and one ECCS recirculation subsystem shall be OPERABLE.

--------------------------- NOTE ---------------------------
An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation.
------------------------------------------------------------

APPLICABILITY: MODE 4.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Required ECCS residual heat removal (RHR) subsystem inoperable. | A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status. | Immediately |
| B. Required ECCS Recirculation subsystem inoperable. | B.1 Restore required ECCS recirculation subsystem to OPERABLE status. | 1 hour |
| C. Required Action and associated Completion Time of Condition B not met. | C.1 Be in MODE 5. | 24 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.3.1 | The following SRs are applicable for all equipment required to be OPERABLE:<br>SR 3.5.2.3 SR 3.5.2.7 | In accordance with applicable SRs |

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.3 ECCS – Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS – Operating," is applicable to these Bases, with the following modifications.

In MODE 4, one ECCS residual heat removal (RHR) subsystem and one ECCS Recirculation subsystem are required.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) or the containment or recirculation sump can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one ECCS residual heat removal (RHR) subsystem and one ECCS Recirculation subsystem are required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36.

LCO

In MODE 4, one ECCS residual heat removal (RHR) subsystem and one ECCS Recirculation subsystem are required to be OPERABLE to

BASES

LCO (continued)

ensure that sufficient ECCS flow is available to the core following a DBA.

In MODE 4, ECCS requirements may be met using containment Recirculation subsystem 31 or 32 and RHR subsystem 31 or 32.

An ECCS RHR subsystem consists of one RHR pump and one RHR heat exchanger as well as associated piping and valves and instrumentation and controls needed to transfer water from the RWST or containment sump to the core. Either RHR heat exchanger may be used with either RHR pump to meet requirements for an RHR subsystem.

A containment Recirculation subsystem consists of one Containment Recirculation pump and one RHR heat exchanger as well as associated piping, valves, instrumentation and controls needed to transfer water from the recirculation sump to the core. Note that Recirculation pump OPERABILITY requires the functional availability of the associated auxiliary component cooling water pump. Either RHR heat exchanger may be used with either recirculation pump to meet requirements for a recirculation subsystem. The same RHR heat exchanger may be used to meet requirements for both the RHR subsystem and the Recirculation subsystem.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the RHR pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, the recirculation flow path using the Recirculation sump or containment sump may be used to deliver its flow to the RCS cold legs.

This LCO is modified by a Note that allows an RHR subsystem to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS residual heat removal (RHR) subsystem and one OPERABLE ECCS Recirculation subsystem is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

BASES

ACTIONS (continued)

B.1

With no containment Recirculation subsystem OPERABLE, due to the inoperability of the pump or flow path from the recirculation sump, the plant is not prepared to provide long term cooling response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS Recirculation subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where a recirculation subsystem is not required.

C.1

Note: Condition C should not be entered if Condition A is applicable.

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

The applicable references from Bases 3.5.2 apply.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.3:
## "ECCS - Shutdown"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.3-1 | 154 | 154 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-2 | 154 | 154 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-14 | 132 | 132 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-15 | 139 TSCR 97-175 | 139 TSCR 97-175 | IPN 97-175 | Changes to Bases Pages | |
| 3.3-17 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-1 | 142 | 142 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-7 | 178 | 178 | No TSCRs | No TSCRs for this Page | N/A |

A.2 A.1

3.3 ENGINEERED SAFETY FEATURES

~~Applicability~~

~~Applies to the operating status of the Engineered Safety Features.~~

~~Objective~~

~~To define those limiting conditions for operating that are necessary: 1) to remove decay heat from the core in emergency or normal shutdown situations; 2) to remove heat from containment in normal operating and emergency situations; 3) to remove airborne iodine from the containment atmosphere following a Design Basis Accident; 4) to minimize containment leakage to the environment subsequent to a Design Basis Accident; 5) to minimize the potential for and consequences of Reactor Coolant System pressure transients.~~

Specification

The following specifications apply except during low temperature physics tests.

LCO 3.5.3 A. Safety Injection and Residual Heat Removal Systems Add Note A.3

Applicability 1. ~~The reactor coolant system $T_{avg}$ shall not exceed 200°F unless the following requirements are met:~~ Mode 4 A.1

SEE ITS 3.5.4 a. The refueling water storage tank water level shall be a minimum of 35.4 feet, with the water at a boron concentration ≥2400 ppm and ≤2600 ppm.

SEE ITS 3.3.3 b. One refueling water storage tank low level alarm operable and set to alarm between 10.5 feet and 12.5 feet of water in the tank.

A.1

subsystem LA.1

LCO 3.5.3 ~~c.~~ One residual heat removal ~~pump and heat exchanger together with the associated piping and valves operable.~~

SEE ITS 3.5.4 ~~d.~~ One recirculation ~~pump together with its associated piping and valves operable.~~

Recirculation L.1

Req. Act B.1 C.1 ~~2.~~ If the ~~Safety Injection and~~ Residual ~~Heat Removal Systems~~ are not restored to meet the requirements of 3.3.A.1 within 1 hour the reactor shall be in the cold shutdown condition within ~~the next~~ ~~20~~ 24 hours. L.2

3. The reactor coolant system $T_{avg}$ shall not exceed 350°F unless the following requirements are met:

SEE ITS 3.5.4

a. The refueling water storage tank water level shall be a minimum of 35.4 feet, with the water at a boron concentration ≥2400 ppm and ≤2600 ppm.

b. DELETED

SEE ITS 3.5.1

c. The four accumulators are pressurized between 600 and 700 psig and each contains a minimum of 775 $ft^3$ and a maximum of 815 $ft^3$ of water at a boron concentration ≥2000 ppm and ≤2600 ppm. Accumulator isolation valves 894A, B, C, and D shall be open and their power supplies deenergized whenever the reactor coolant system pressure is above 1000 psig.

Add Condition A and Req. Act A.1 L.1

Add SR 3.5.3.1 L.3

A.1

Bases

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature, by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.[1] With this mode of startup, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation, and, therefore, the minimum required engineered safeguards and auxiliary cooling systems are required to be operable.

The probability of sustaining both a major accident and a simultaneous failure of a safeguards component to operate as designed is necessarily very small. Thus, operation with the reactor above the cold shutdown condition with minimum safeguards operable for a limited period does not significantly increase the probability of an accident having consequences which are more severe than the Design Basis Accident.

The operable status of the various systems and components is demonstrated by periodic tests defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible, in most cases, to effect repairs and restore the system to full operability within a relatively short time. The inoperability of a single component does not negate the ability of the system to perform its function,[2] but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. Assurance that the redundant component(s) will operate if required to do so exists if the required periodic surveillance testing is current and there are no known reasons to suggest that the redundant component(s) are inoperable. If it develops that (a) the inoperable component is not repaired within the specified allowable time period, or (b) a second component in the same or related system is found to be inoperable, the reactor, if critical, will initially be brought to the hot shutdown condition utilizing normal operating procedures to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. If the reactor was already subcritical, the reactor coolant system temperature and pressure will be maintained within the stated values in order to limit the amount of stored energy in the reactor coolant system. The stated tolerances provide a band for operator control. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the

cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would damage the fuel elements or result in a release in excess of 10 CFR 100 and 10 CFR 50 dose limits.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and, therefore, in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are based on two considerations:

1) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full rated power, the magnitude of the decay heat decreases after initiating hot shutdown. Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 1 hour of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case the reactor is to be put into the cold shutdown condition.

The limits for the Refueling Water Storage Tank and the accumulators insure the required amount of water with the proper boron concentration for injection into the reactor coolant system following a loss-of-coolant accident is available. These limits are based on values used in the accident analysis.(9)(13)

A.1

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.[3] The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of their performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met.[1]

During operating modes in the temperature range between 200°F and 350°F, a sufficient decay heat removal capability is provided by a reactor coolant pump with a steam generator heat sink or a residual heat removal loop. This redundancy ensures that a single failure will not result in a complete loss of decay heat removal. Above 350°F, the normal RHR suction line is isolated from the RCS to protect RHR piping from overpressurization due to inadvertent SI pump actuation.

During operating modes when the reactor coolant $T_{avg}$ is less than 200°F, but not in the refueling operation condition, a sufficient decay heat removal capability is provided by a residual heat removal loop.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) containment spray with sodium hydroxide addition. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 95°F).[4] In the event of a Design Basis Accident, any one of the following configurations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting off-site doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also in the event of a Design Basis Accident, any one of three configurations of fan-cooler units (with charcoal filters) and/or containment spray pumps (with sodium hydroxide addition) will reduce airborne organic and molecular iodine activities sufficiently to limit off-site doses to acceptable values.[5] Any one of these three configurations constitutes the minimum safeguards for iodine removal.

The combination of three fan-coolers and one containment spray pump is capable of being operated on emergency power with one diesel generator failing to start. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of off-site power or operation of all emergency diesel generators.

A.1

A.1

## 4.5 TESTS FOR ENGINEERED SAFETY FEATURES AND AIR FILTRATION SYSTEMS

A.2

~~Applicability~~

~~Applies to testing of the Safety Injection System, the Containment Spray System, the Hydrogen Recombiner System, and the Air Filtration Systems.~~

~~Objective~~

~~To verify that the subject systems will respond promptly and perform their design functions, if required.~~

~~Specification~~

A. SYSTEM TESTS

1. Safety Injection System

LCO 3.5.3

a. ~~System tests shall be performed at least once per 24 months*. With the Reactor Coolant System pressure less than or equal to 350 psig and temperature less than or equal to 350°F, a test safety injection signal will be applied to initiate operation of the system. The safety injection and residual heat removal pumps are made inoperable for this test.~~ L3

b. ~~The test will be considered satisfactory if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing, that is, the appropriate pump breakers shall have opened and closed, and the appropriate valves shall have completed their travel.~~ L3

SEE ITS 3.5.2

c. Conduct a flow test of the high head safety injection system after any modification is made to either its piping and/or valve arrangement.

d. Verify that the mechanical stops on Valves 856 A, C, D, E, F, H, J and K are set at the position measured and recorded during the most recent ECCS operational flow test or flow tests performed in accordance with (c) above. This surveillance procedure shall be performed following any maintenance on these valves or their associated motor operators and at a convenient outage if the position of the mechanical stops have not been verified in the preceding three months.

SEE ITS 3.8.1

* The time delay relays will be tested at intervals no greater than 22.5 months (18 months + 25%).

B. Component Tests

Add SR 3.5.3.1 (SR 3.5.2.7) — M.1

1. Pumps

SR 3.5.3.1 (SEE ITS 3.5.2 for SR 3.5.2.3)

a. The safety injection pumps, residual heat removal pumps, containment spray pumps and the auxiliary component cooling water pumps shall be started at intervals not greater than one month. The recirculation pumps shall be started at least once per 24 months.

b. Acceptable levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least fifteen minutes.

2. Valves

SEE ITS 3.6.7

a. Each spray additive valve shall be cycled by operator action with the pumps shut down at least once per 24 months.

SEE ITS 3.5.1

b. The accumulator check valves shall be checked for operability at least once per 24 months.

SEE ITS 3.4.14

c. The following check valves shall be checked for gross leakage at least once per 24 months:

| | | | |
|---|---|---|---|
| 857A & G | 857J | 857S & T | 897B |
| 857B | 857K | 857U & W | 897C |
| 857C | 857L | 895A | 897D |
| 857D | 857M | 895B | 838A |
| 857E | 857N | 895C | 838B |
| 857F | 857P | 895D | 838C |
| 857H | 857Q & R | 897A | 838D |

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.3:
## "ECCS - Shutdown"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.5.3 - ECCS - Shutdown

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety because neither are required by 10 CFR 50.36, and neither define nor impose any specific requirements.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.3.A.1.c requires at least one residual heat removal (RHR) subsystem operable for ECCS injection when in Mode 4. ITS LCO 3.5.3 maintains the requirement to have at least one RHR subsystem operable for ECCS injection when in Mode 4; however, ITS LCO 3.5.3 is modified by a note that allows an RHR subsystem to be considered Operable for the

ECCS initiation function during alignment and operation for decay heat removal if the RHR subsystem is capable of being manually realigned (remote or local) to the ECCS mode of operation and is not otherwise inoperable. Although this allowance is not specifically stated in CTS 3.3, the requirement for an RHR pump to satisfy the ECCS function in CTS 3.3.A.1.c with a concurrent requirement in CTS 3.3.A.6.a for two RHR pumps in decay heat removal function implies that an RHR pump can satisfy both requirements concurrently. Additionally, CTS does not require the Operability of ECCS automatic initiation functions in Mode 4. Therefore, consistent with industry practice, IP3 does allow an RHR pump to satisfy concurrent requirements for ECCS injection function and decay heat removal function.

This allowance is acceptable because of the stable conditions associated with operation in Mode 4, the reduced probability of occurrence of a Design Basis Accident (DBA)in Mode 4 and the limited core cooling requirements in Mode 4. Therefore, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA in Mode 4.

Adding a statement that an RHR subsystem is Operable for the ECCS initiation function during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable is an administrative change with no impact on safety (See ITS 3.5.3, DOC L.3).

MORE RESTRICTIVE

M.1 CTS 3.3 and CTS 4.5.A do not establish any requirements for the periodic verification that containment sump and recirculation sump suction inlets are unrestricted and otherwise in proper operating condition.

ITS SR 3.5.3.1 is added ( in conjunction with ITS SR 3.5.2.7) to require verification every 24 months that containment sump suction inlets are unrestricted and otherwise in proper operating condition. This Frequency is consistent with the need to perform this verification while the plant is shutdown and, based on industry experience, is sufficient to detect abnormal degradation.

This change is acceptable because it does not introduce any operation that is un-analyzed while requiring periodic verification that analysis assumptions regarding the Operability of ECCS system flow paths are satisfied. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.3.A.2 requires the plant in cold shutdown (Mode 5) if an RHR subsystem is not restored to Operable within 1 hour when in Mode 4. ITS LCO 3.5.3, Required Action B.1, maintains the requirement to place the plant in Mode 5 if a recirculation subsystem is not restored to Operable within 1 hour. However, if a required RHR subsystem is not Operable in Mode 4, ITS LCO 3.5.3, Required Action A.1, requires immediate action to restore an RHR subsystem to Operable but the requirement to place the plant in Mode 5 is eliminated.

This change is needed because if no RHR subsystem is available to perform the ECCS function then it is likely that no RHR subsystem is available to perform the decay heat removal function needed to proceed to Mode 5. Additionally, there is no prohibition against proceeding to Mode 5 but remaining in Mode 4 permits plant temperatures to be maintained such that decay heat removal and plant temperature control can be accomplished using the steam generators.

This change is acceptable because of the stable conditions associated with operation in Mode 4, the low probability of occurrence of a Design Basis Accident (DBA) at the associated pressures and temperatures in Mode 4 and the limited core cooling requirements in Mode 4. Additionally, ITS LCO 3.5.3, Required Action A.1, requires expeditious action to restore the required ECCS RHR injection and decay heat removal capability. Therefore, this change has no significant adverse impact on safety.

L.2 CTS 3.3.A.2 requires the plant be in cold shutdown (Mode 5) within 20 hours if an RHR subsystem (See ITS 3.5.3, DOC L.1) and/or a recirculation subsystem are not restored to Operable within 1 hour when in Mode 4 . ITS LCO 3.5.3, Required Action C.1, maintains the requirement to place the plant in Mode 5 if a recirculation subsystem is not restored to Operable within 1 hour; however, ITS LCO 3.5.3, Required Action C.1, extends the time allowed to reach Mode 5 from 20 hours to 24 hours. This change is needed and is acceptable because, based on operating experience, Mode 4 to Mode 5 in 24 hours is a reasonable time to reach the required plant conditions in an orderly manner and without challenging plant systems. ITS 3.5.3, Required Action C.1, still requires that the plant is promptly placed outside the LCO Applicability when requirements are not met. Therefore, this change has no significant adverse impact on safety.

L.3 CTS 4.5.A and CTS 4.5.B do not differentiate between surveillance test requirements for Operability in Modes 1, 2 and 3, when automatic ECCS initiation is required, and requirements for Operability in Mode 4, when automatic ECCS initiation is not required. ITS SR 3.5.3.1 establishes surveillance test requirements for Operability in Modes 4 which recognize that manual alignment to restore the ECCS function of RHR when in Mode 4 is acceptable (See ITS 3.5.3, DOC A.3 and M.1). Specifically, CTS 4.5.A.1.a and b (ECCS automatic initiation) is not required by ITS SR 3.5.3.1 for operability of ECCS RHR in Mode 4.

This change is needed and is acceptable because ITS LCO 3.5.3 is modified by a note that allows an RHR subsystem to be considered Operable for the ECCS initiation function during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable (See ITS 3.5.3, DOC A.3). Additionally, CTS does not require the Operability of ECCS automatic initiation functions in Mode 4. Therefore, surveillance tests that demonstrate Operability of ECCS RHR automatic initiation and/or valve positioning are not required in Mode 4. Dependence on ECCS RHR being manually realigned (remote or local) to the ECCS mode of operation is acceptable because of the stable conditions associated with operation in Mode 4, the reduced probability of occurrence of a Design Basis Accident (DBA) and the limited core cooling requirements. In Mode 4, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

LA.1 CTS 3.3.A.3.e, f, and g require Operability of three safety injection pumps, two residual heat removal pumps and heat exchangers, and two recirculation pumps with the associated piping and valves whenever the reactor is above 350°F (Modes 1, 2 and 3).

ITS 3.5.3 requires Operability of three ECCS trains with the ECCS trains defined in the ITS 3.5.2 Bases and system descriptions in the FSAR. Specifically, the ITS 3.5.2 Bases specify that the ECCS Function is provided by three separate ECCS systems: high head safety injection (HHSI), residual heat removal (RHR) injection, and containment recirculation. Each ECCS system is divided into subsystems as follows: three 50% capacity HHSI subsystems; two 100% capacity RHR subsystems; and, two 100% capacity recirculation subsystems. Each of these subsystems is further described as including the required valves, heat exchangers and flow paths needed for the subsystem to perform its safety function. Furthermore, the ECCS subsystems (3 HHSI, 2 RHR and 2 Recirculation) are grouped into three trains (5A, 2A/3A and 6A) such that any two of the 3 trains are capable of meeting all ECCS capability assumed in the accident analysis. The ITS 3.5.2 ECCS trains use the same designation as the Safeguards Power Trains required by LCO 3.8.9, Distribution Systems - Operating, with Safeguards Power Train 5A supported by DG 33, Safeguards Power Train 2A/23 supported by DG 31, Safeguards Power Train 6A supported by DG 32.

Establishing ECCS requirements in terms of ECCS trains with the ECCS subsystems and trains defined in the ITS 3.5.2 Bases is needed because this presentation ensures that requirements are clearly understood and consistently applied in conjunction with ITS 3.8.1, AC Sources - Operating, and ITS 3.8.9, Distribution Systems - Operating.

This change is acceptable because ITS 3.5.3 maintains the existing requirement for the Operability of three trains of ECCS; therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

This change, which allows the description of the design of the ECCS systems to be maintained in the FSAR and the detailed description of the requirements for Operability of these systems to be maintained in the ITS Bases, is consistent with the approach used in NUREG-1431 for all Limiting Conditions for Operation (LCOs). This approach is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, and ITS 5.5.13, Technical Specifications (TS) Bases Control Program, are designed to assure that changes to the FSAR and ITS Bases do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 and ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of FSAR and Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.3:
## "ECCS - Shutdown"

## PART 4:

## No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

# NO SIGNIFICANT HAZARDS EVALUATION
## ITS SECTION 3.5.3 - ECCS - Shutdown

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change eliminates the requirement to proceed to Mode 5 if the one required ECCS RHR subsystem is inoperable when in Mode 4; ITS LCO 3.5.3, Required Action A.1, requires only that action to restore an RHR subsystem to Operable is initiated immediately.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the Operability status of RHR systems when in Mode 4 has no affect on the initiators of any analyzed events (assuming alternate decay heat removal capability is available).

   This change will not result in a significant increase in the consequences of an accident previously evaluated because if no RHR subsystem is available to perform the ECCS function then it is likely that no RHR subsystem is available to perform the decay heat removal function needed to proceed to Mode 5 expeditiously. Additionally, there is no prohibition against preceding to Mode 5 but remaining in Mode 4 permits plant temperatures to be maintained so that decay heat removal and temperature control can be accomplished using the steam generators.

   This change is acceptable because of the stable conditions associated with operation in Mode 4, the low probability of occurrence of a Design Basis Accident (DBA) at the associated pressures and temperatures and the limited core cooling requirements. Additionally, ITS LCO 3.5.3, Required Action A.1, requires expeditious action to restore the required ECCS RHR injection and decay heat removal capability.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS systems are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because if no RHR subsystem is available to perform the ECCS function then it is likely that no RHR subsystem is available to perform the decay heat removal function needed to proceed to Mode 5 expeditiously. Additionally, there is no prohibition against preceding to Mode 5 but remaining in Mode 4 permits plant temperatures to be maintained such that decay heat removal and plant temperature control can be accomplished using the steam generators.

   This change is acceptable because of the stable conditions associated with operation in Mode 4, the low probability of occurrence of a Design Basis Accident (DBA) at the associated pressures and temperatures and the limited core cooling requirements. Additionally, ITS LCO 3.5.3, Required Action A.1, requires expeditious action to restore the required ECCS RHR injection and decay heat removal capability.

---

LESS RESTRICTIVE
<u>("L.2" Labeled Comments/Discussions)</u>

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.5.3 - ECCS - Shutdown

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change extends the time allowed to reach Mode 5 from Mode 4 when an a recirculation subsystem is not restored to Operable within 1 hour from 20 hours to 24 hours.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the Operability status of ECCS systems has no affect on the initiators of any analyzed events. This change will not result in a significant increase in the consequences of an accident previously evaluated because, based on operating experience, Mode 4 to Mode 5 in 24 hours is a reasonable time to reach the required plant conditions in an orderly manner and without challenging plant systems. ITS 3.5.3, Required Action C.1, still requires that the plant is promptly placed outside the LCO Applicability when requirements are not met.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS systems are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because, based on operating experience, Mode 4 to Mode 5 in 24 hours is a reasonable time to reach the required plant conditions in an orderly manner and without challenging plant systems. ITS 3.5.3, Required Action C.1, still requires that the plant is promptly placed outside the LCO Applicability when requirements are not met.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.5.3 - ECCS - Shutdown

LESS RESTRICTIVE
("L.3" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change establishes surveillance test requirements for Operability in Modes 4 which recognize that manual alignment to restore the ECCS function of RHR when in Mode 4 is acceptable. Specifically, CTS 4.5.A.1.a and b (ECCS automatic initiation) is not required for Operability of ECCS RHR in Mode 4.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the Operability status of ECCS systems has no affect on the initiators of any analyzed events.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because ITS LCO 3.5.3 is modified by a note that allows an RHR subsystem to be considered Operable for the ECCS initiation function during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. Additionally, CTS does not require the Operability of ECCS automatic initiation functions in Mode 4. Therefore, surveillance tests that demonstrate Operability of ECCS RHR automatic initiation and/or valve positioning are not required in Mode 4. Dependence on ECCS RHR being manually realigned (remote or local) to the ECCS mode of operation is acceptable because of the stable conditions associated with operation in Mode 4, the reduced probability of occurrence of a Design Basis Accident (DBA) and the limited core cooling requirements. In Mode 4, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way ECCS systems are operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because ITS LCO 3.5.3 is modified by a note that allows an RHR subsystem to be considered Operable for the ECCS initiation function during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. Additionally, CTS does not require the Operability of ECCS automatic initiation functions in Mode 4. Therefore, surveillance tests that demonstrate Operability of ECCS RHR automatic initiation and/or valve positioning are not required in Mode 4. Dependence on ECCS RHR being manually realigned (remote or local) to the ECCS mode of operation is acceptable because of the stable conditions associated with operation in Mode 4, the reduced probability of occurrence of a Design Basis Accident (DBA) and the limited core cooling requirements. In Mode 4, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.5.3:
# "ECCS - Shutdown"

# PART 5:

# NUREG-1431
# Annotated to show differences between
# NUREG-1431 and ITS

## Status of NUREG 1431 Generic Changes for ITS 3.5.3

This ITS Specification is based on NUREG-1431 Specification No. **3.5.3**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| WOG-016 | **090 R0** | ADD A NOTE TO LCO 3.5.3 THAT ALLOWS RHR TO BE OPERABLE AS ECCS WHEN ALIGNED FOR DECAY HEAT REMOVAL | See Next Rev. | See R1 | **N/A** |
| WOG-016 R1 | **090 R1** | ADD A NOTE TO LCO 3.5.3 THAT ALLOWS RHR TO BE OPERABLE AS ECCS WHEN ALIGNED FOR DECAY HEAT REMOVAL | Approved by NRC | Incorporated | **T.1** |

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.3 ECCS—Shutdown

LCO 3.5.3 ~~One ECCS train~~ shall be OPERABLE. Insert: 3.5-7-01 ← Insert: From Page 3.5-8 T.1

CLB.1

⟨3.3.A.1⟩ ⟨DOC A.3⟩ APPLICABILITY: MODE 4.

ACTIONS

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| ⟨3.3.A.2⟩ ⟨DOC L.1⟩ | [A. Required ECCS residual heat removal (RHR) subsystem inoperable. | A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status. | Immediately] |
| ⟨3.3.A.2⟩ | B. Required ECCS ~~[high head]~~ Recirculation subsystem inoperable. | B.1 Restore required ECCS ~~[high head]~~ Recirculation subsystem to OPERABLE status. | 1 hour |
| ⟨3.3.A.2⟩ ⟨DOC L.2⟩ | C. Required Action and associated Completion Time [of Condition B] not met. | C.1 Be in MODE 5. | 24 hours |

NUREG-1431 Markup Inserts
ITS SECTION 3.5.3 - ECCS - Shutdown

**Insert: 3.5-7-01**:

One ECCS residual heat removal (RHR) subsystem and one ECCS recirculation subsystem CLB.1

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.3.1 | ---NOTE--- An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation. ------<br><br>The following SRs are applicable for all equipment required to be OPERABLE:<br><br>~~SR 3.5.2.1~~ ~~SR 3.5.2.7~~<br>~~SR 3.5.2.3~~ SR 3.5.2.~~8~~ 7<br>SR 3.5.2.~~4~~ 3 | In accordance with applicable SRs |

T.1

<4.5.B.1>
<4.5.A.1.d>
<DOC L.3>
<DOC M.1>

Move Note Page 3.5-7

NUREG-1431 Markup Inserts
ITS SECTION 3.5.3 - ECCS - Shutdown

Insert: B 3.5-21-01:

one ECCS residual heat removal (RHR) subsystem and one ECCS recirculation subsystem are required.

Insert: B 3.5-21-02:

, containment sump, and recirculation sump

Insert: B 3.5-21-03:

one ECCS residual heat removal (RHR) subsystem and one ECCS Recirculation subsystem are

Insert: B 3.5-21-04:

one ECCS residual heat removal (RHR) subsystem and one ECCS Recirculation subsystem are

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.3 ECCS—Shutdown

### BASES

**BACKGROUND**

The Background section for Bases 3.5.2, "ECCS—Operating," is applicable to these Bases, with the following modifications.

In MODE 4, ~~the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head)~~. DB.1

Insert: B3.5-21-01

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

Insert: B3.5-21-02

**APPLICABLE SAFETY ANALYSES**

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that ~~contain~~ automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Insert B3.5-21-03

Only ~~one train of ECCS is~~ required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS ~~trains~~ satisfy Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36

**LCO**

In MODE 4, ~~one of the two independent (and redundant) ECCS trains is~~ required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

Insert: B3.5-21-04

In MODE 4, ~~an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow~~

Insert: B3.5-21-05

(continued)

**Insert: B 3.5-21-05**:

ECCS requirements may be met using containment recirculation subsystem 31 or 32 and RHR subsystem 31 or 32.

An ECCS RHR subsystem consists of one RHR pump and one RHR heat exchanger as well as associated piping and valves and instrumentation and controls needed to transfer water from the RWST to the core. Either RHR heat exchanger may be used with either RHR pump to meet requirements for an RHR heat exchanger.

A containment recirculation subsystem consists of one Containment Recirculation pump and one RHR heat exchanger as well as associated piping and valves and instrumentation and controls needed to transfer water from the recirculation sump to the core. Note that Recirculation pump OPERABILITY requires the functional availability of the associated auxiliary component cooling water pump. Either RHR heat exchanger may be used with either recirculation pump to meet requirements for an RHR heat exchanger. The same RHR heat exchanger may be used to meet requirements for both the RHR subsystem and the recirculation subsystem.

BASES

LCO (continued)

~~path capable of taking suction from the RWST and transferring suction to the containment sump.~~

RHR

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS ~~hot and~~ cold legs.

recirculation sump or

Insert: B 3.5-22-01

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, ~~one OPERABLE ECCS train~~ is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

Insert: B3.5-22-02

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.~~5~~ 4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.~~6~~ 5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must

(continued)

INSERT B 3.5-22-01:

T.1

This LCO is modified by a Note that allows an RHR subsystem to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

INSERT B 3.5-22-02:

one OPERABLE ECCS residual heat removal (RHR) subsystem and one OPERABLE ECCS recirculation subsystem

BASES

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ACTIONS

A.1 (continued)

continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

B.1

containment recirculation

With no ~~ECCS high head~~ subsystem OPERABLE, due to the inoperability of the ~~centrifugal charging~~ pump or flow path from the ~~RWST~~, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ~~ECCS high head~~ subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

recirculation sump

long term cooling

Recirculation subsystem

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

Insert: B 3.5-23-01

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SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply. ~~This SR is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of~~

T.1

(continued)

---

BASES

**SURVEILLANCE REQUIREMENTS**

SR 3.5.3.1 (continued)

~~operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4, if necessary.~~ T.1

**REFERENCES**

The applicable references from Bases 3.5.2 apply.

INSERT B 3.5-23-01:

Required Action C.1 does not mandate a cooldown to MODE 5 when a required ECCS RHR subsystem is not OPERABLE (i.e., Condition A) because plant cooldown may not be possible with inoperable RHR subsystems.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.3:
## "ECCS - Shutdown"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.5.3 - ECCS - Shutdown

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 NUREG 1431, Rev. 1, requires that one ECCS train shall Operable in Mode 4. IP3 ITS LCO 3.5.3 requires one ECCS residual heat removal (RHR) subsystem and one ECCS recirculation subsystem Operable in Mode 4. This change is needed and is acceptable because the ECCS residual heat removal (RHR) subsystem and one ECCS recirculation subsystem supply sufficient volume at the required pressure to provide adequate makeup in the event of a LOCA in Mode 4. This difference is consistent with the current licensing basis because CTS 3.3.A does not require the safety injection (high head) pumps to be operable in Mode 4.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical errors or made a minor editorial improvements to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, these changes are not significant or generic deviations from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-90, Rev.1 (WOG-12) which adds a Note that allows RHR to be considered Operable as an ECCS system when aligned for decay heat removal. This is an approved generic change traveler for NUREG-1431.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.5.4:
# "Refueling Water Storage Tank (RWST)"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. RWST boron concentration not within limits of SR 3.5.4.3.<br><br><u>OR</u><br><br>RWST borated water temperature not within limits of SR 3.5.4.1. | A.1 Restore RWST to OPERABLE status. | 8 hours |
| B. RWST inoperable for reasons other than Condition A. | B.1 Restore RWST to OPERABLE status. | 1 hour |
| C. Required Action and associated Completion Time not met. | C.1 Be in MODE 3.<br><br><u>AND</u><br><br>C.2 Be in MODE 5. | 6 hours<br><br><br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.4.1 | -----------------NOTE------------------<br>Only required to be performed when ambient air temperature remains < 40°F or > 100°F for 24 hours.<br>---------------------------------------<br><br>Verify RWST borated water temperature is ≥ 40°F and ≤ 110°F. | 24 hours |
| SR 3.5.4.2 | Verify RWST borated water level is ≥ 35.4 feet. | 7 days |
| SR 3.5.4.3 | Verify RWST boron concentration is ≥ 2400 ppm and ≤ 2600 ppm. | 31 days |

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling cavity during refueling, to the ECCS to fill accumulators, and to the ECCS and the Containment Spray System during accident conditions.

The RWST supplies the ECCS and the Containment Spray System through separate supply headers during the injection phase of a loss of coolant accident (LOCA). Motor operated isolation valves are provided to isolate the RWST from the ECCS subsystems once the system has been transferred to the recirculation mode. The switchover to the cold leg recirculation phase is manually initiated when the RWST level has reached the low-low alarm setpoint and sufficient coolant inventory to support pump operation in recirculation mode is verified to be in the containment. Use of a single RWST to supply all of the injection trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

During normal operation in MODES 1, 2, and 3, the high head safety injection (HHSI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

The ECCS and Containment Spray System pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the recirculation sump or the containment sump to support continued operation of the

BACKGROUND (continued)

ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA or MSLB.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment due to improper pH in the sumps.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating"; B 3.5.3, "ECCS - Shutdown"; and B 3.6.6, "Containment Spray System and Containment Fan Fooler System." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the open analyses.

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered.

APPLICABLE SAFETY ANALYSES (continued)

For a large break LOCA analysis, the minimum water volume limit of 195,800 gallons and the lower boron concentration limit of 2300 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 2600 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 40°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of 110°F is used in the LOCA containment integrity analysis. Exceeding this temperature will result in higher containment pressures due to reduced containment spray cooling capacity. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST satisfies Criterion 3 of 10 CFR 50.36.

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the recirculation sump and the containment sump to support ECCS pump operation in the recirculation mode.

LCO (continued)

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level."

ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits of SR 3.5.4.3 and SR 3.5.4.1, respectively, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place

BASES

ACTIONS

B.1 (continued)

the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance unless ambient air temperatures are not within the operating limits of the RWST for more than 24 hours. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.

SR 3.5.4.2

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.3

The boron concentration of the RWST should be verified every 31 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST level is normally stable, a 31 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR, Chapter 6 and Chapter 14.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.4:
## "Refueling Water Storage Tank (RWST)"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.3-1 | 154 | 154 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-2 | 154 | 154 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-5 | 53 | 53 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-14 | 132 | 132 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-15 | 139 TSCR 97-175 | 139 TSCR 97-175 | IPN 97-175 | Changes to Bases Pages | |
| 3.3-16 | 154 | 154 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-17 | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| T 4.1-2(1) | 139 | 139 | No TSCRs | No TSCRs for this Page | N/A |

3.3 ENGINEERED SAFETY FEATURES

A.2

~~**Applicability**~~

~~Applies to the operating status of the Engineered Safety Features.~~

~~**Objective**~~

~~To define those limiting conditions for operating that are necessary: 1) to remove decay heat from the core in emergency or normal shutdown situations; 2) to remove heat from containment in normal operating and emergency situations; 3) to remove airborne iodine from the containment atmosphere following a Design Basis Accident; 4) to minimize containment leakage to the environment subsequent to a Design Basis Accident; 5) to minimize the potential for and consequences of Reactor Coolant System pressure transients.~~

**Specification**

The following specifications apply except during low temperature physics tests.

A. Safety Injection and Residual Heat Removal Systems Mode 1,2,3 and 4 A.3

LCO 3.5.4 Applicability

1. ~~The reactor coolant system $T_{avg}$ shall not exceed 200°F unless the following requirements are met:~~

LCO 3.5.4 SR 3.5.4.2 SR 3.5.4.3

a. The refueling water storage tank water level shall be a minimum of 35.4 feet, with the water at a boron concentration ≥2400 ppm and ≤2600 ppm.

SEE ITS 3.3.3

b. One refueling water storage tank low level alarm operable and set to alarm between 10.5 feet and 12.5 feet of water in the tank.

Add SR 3.5.4.1 M.1

Add SR 3.5.4.2 M.4

Add Condition A and associated Req. Act

M.

L.2

SEE ITS 3.5.3

c. One residual heat removal pump and heat exchanger together with the associated piping and valves operable.

d. One recirculation pump together with its associated piping and valves operable.

Req. Act. B.1 C.1 C.2

~~2.~~ If the ~~Safety Injection and Residual Heat Removal Systems are~~ RWST not restored to meet the requirements of 3.3.A.1 within 1 hour the reactor shall be in the cold shutdown condition within ~~the next~~ ~~20~~ 36 hours. L.1

LCO 3.5.4 Applicability

3. ~~The reactor coolant system $T_{avg}$ shall not exceed 350°F unless the following requirements are met:~~ Mode 1, 2, 3 and 4. A.3

LCO 3.5.4 SR 3.5.4.2 SR 3.5.4.3

a. The refueling water storage tank water level shall be a minimum of 35.4 feet, with the water at a boron concentration ≥2400 ppm and ≤2600 ppm.

b. DELETED

SEE ITS 3.5.1

c. The four accumulators are pressurized between 600 and 700 psig and each contains a minimum of 775 $ft^3$ and a maximum of 815 $ft^3$ of water at a boron concentration ≥2000 ppm and ≤2600 ppm. Accumulator isolation valves 894A, B, C, and D shall be open and their power supplies deenergized whenever the reactor coolant system pressure is above 1000 psig.

Add Condition A and associated Req. Act M.1 L.2

RWST

5. If the ~~Safety Injection and Residual Heat Removal Systems are~~ not restored to meet the requirements ~~of 3.3.A.3~~ within the time periods specified in 3.3.A.4; then: M.2

a. ~~If~~ the reactor ~~is critical, it~~ shall be in the hot shutdown condition within four 6 hours and the cold shutdown condition within the following 24 36 hours. L.3

b. ~~If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.A.3 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition using normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.~~ M.2

6. When the reactor coolant system $T_{avg}$ is greater than 200°F and less than 350°F, the following decay heat removal requirements shall be met:

a. Two residual heat removal pumps together with their associated heat exchangers, piping, and valves shall be operable,

OR

b. A minimum of one residual heat removal pump and heat exchanger and a minimum of one reactor coolant pump and steam generator together with their associated piping and valves, shall be operable,

OR

SEE ITS 3.4.6 3.4.7 3.4.8

c. A minimum of two reactor coolant pumps and two steam generators, together with their associated piping and valves, shall be operable,

OR

d. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system. Otherwise, if sufficient equipment is available, be in cold shutdown within 20 hours.

7. When the reactor coolant $T_{avg}$ is less than 200°F, but not in the refueling operation condition, two residual heat removal pumps, together with their associated heat exchangers, piping and valves, shall be operable.

a. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system.

b. The above requirements may be suspended during maintenance, modifications, testing, inspection or repair provided that:

1) an alternate means of decay heat removal is available and return of the system within sufficient time to prevent exceeding cold shutdown requirements is assured;

Bases

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature, by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.[1] With this mode of startup, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation, and, therefore, the minimum required engineered safeguards and auxiliary cooling systems are required to be operable.

The probability of sustaining both a major accident and a simultaneous failure of a safeguards component to operate as designed is necessarily very small. Thus, operation with the reactor above the cold shutdown condition with minimum safeguards operable for a limited period does not significantly increase the probability of an accident having consequences which are more severe than the Design Basis Accident.

A.1

The operable status of the various systems and components is demonstrated by periodic tests defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible, in most cases, to effect repairs and restore the system to full operability within a relatively short time. The inoperability of a single component does not negate the ability of the system to perform its function,[2] but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. Assurance that the redundant component(s) will operate if required to do so exists if the required periodic surveillance testing is current and there are no known reasons to suggest that the redundant component(s) are inoperable. If it develops that (a) the inoperable component is not repaired within the specified allowable time period, or (b) a second component in the same or related system is found to be inoperable, the reactor, if critical, will initially be brought to the hot shutdown condition utilizing normal operating procedures to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. If the reactor was already subcritical, the reactor coolant system temperature and pressure will be maintained within the stated values in order to limit the amount of stored energy in the reactor coolant system. The stated tolerances provide a band for operator control. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the

cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would damage the fuel elements or result in a release in excess of 10 CFR 100 and 10 CFR 50 dose limits.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and, therefore, in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are based on two considerations:

1) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full rated power, the magnitude of the decay heat decreases after initiating hot shutdown. Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 1 hour of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case the reactor is to be put into the cold shutdown condition.

The limits for the Refueling Water Storage Tank and the accumulators insure the required amount of water with the proper boron concentration for injection into the reactor coolant system following a loss-of-coolant accident is available. These limits are based on values used in the accident analysis.(9)(13)

A.1

A.1

The minimum indicated RWST level of 35.4 feet (approximately 342,200 gals.) and the low level alarms ("allowable values") of 10.5 feet (approx. 111,100 gals.) and 12.5 feet (approx. 129,700 gals.), include consideration for instrumentation uncertainties, margin, and the unusable volume at the bottom of the tank.[17][18] These water levels ensure a minimum of approx. 195,800 gals. available for injection, and approx. 66,700 gals. for use during and following the transition from injection to recirculation (to allow continued CS pump operation for sump pH control).[18] The minimum RWST boron concentration ensures that the reactor core will remain subcritical during long term recirculation with all control rods fully withdrawn following a postulated large break LOCA.

The four accumulator isolation valves (894 A,B,C,D) are maintained in the open position when the reactor coolant pressure is above 1000 psig to assure flow passage from the accumulators will be available during the injection phases of a loss-of-coolant accident. Indication is also provided on the monitor light panel, should any of these valves not be in the full open position even with the valve operator deenergized. The 1000 psig limit is derived from the minimum pressure requirements of the accumulators combined with instrument error and an operational band and is based upon avoiding inadvertent injection into the reactor coolant system. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occurring when accumulator core cooling flow is required. Valves 856 B and G are maintained in the closed position to prevent hot leg injection during the injection phase of a loss-of-coolant accident. As an additional assurance of preventing hot leg injection, these valve motor operators are deenergized to prevent spurious opening of these valves during the injection phase of a loss-of-coolant accident. Power will be restored to these valves at an appropriate time in accordance with plant operating procedures after a loss-of-coolant accident in order to establish hot leg recirculation.

Valves 1810, 882, and 744 are maintained in the open position to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As additional assurance of flow passage availability, these valve motor operators are de-energized to prevent an extremely unlikely spurious closure. This additional precaution is acceptable, since failure to manually re-establish power to close these valves following the injection phase is tolerable as a single failure.

Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the refueling water storage tank are de-energized in the open position to prevent an extremely unlikely spurious closure which would cause the safety injection pumps to overheat if the reactor coolant system pressure is above the shutoff head of the pumps.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.[3] The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of their performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met.[1]

During operating modes in the temperature range between 200°F and 350°F, a sufficient decay heat removal capability is provided by a reactor coolant pump with a steam generator heat sink or a residual heat removal loop. This redundancy ensures that a single failure will not result in a complete loss of decay heat removal. Above 350°F, the normal RHR suction line is isolated from the RCS to protect RHR piping from overpressurization due to inadvertent SI pump actuation.

During operating modes when the reactor coolant $T_{avg}$ is less than 200°F, but not in the refueling operation condition, a sufficient decay heat removal capability is provided by a residual heat removal loop.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) containment spray with sodium hydroxide addition. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 95°F).[4] In the event of a Design Basis Accident, any one of the following configurations will provide sufficient cooling to reduce containment pressure at a rate consistent with limiting off-site doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also in the event of a Design Basis Accident, any one of three configurations of fan-cooler units (with charcoal filters) and/or containment spray pumps (with sodium hydroxide addition) will reduce airborne organic and molecular iodine activities sufficiently to limit off-site doses to acceptable values.[5] Any one of these three configurations constitutes the minimum safeguards for iodine removal.

The combination of three fan-coolers and one containment spray pump is capable of being operated on emergency power with one diesel generator failing to start. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of off-site power or operation of all emergency diesel generators.

A.1

TABLE 4.1-2 (Sheet 1 of 2)

| FREQUENCIES FOR SAMPLING TESTS | | | |
|---|---|---|---|
| Sample | Analysis | Frequency | Maximum Time Between Analysis |
| 1. Reactor Coolant | Gross Activity(1) | 5 days/week(1)(4) | 3 days(4) |
| | Tritium Activity | Weekly(1) | 10 days |
| | Boron concentration | 2 days/week | 5 days |
| | Radiochemical (gamma)(2) Spectral Check | Monthly | 45 days |
| | Oxygen and Chlorides Concentration | 3 times per 7 days | 3 days |
| | Fluorides Concentration | Weekly | 10 days |
| | Ē Determination (3) | Semi-Annually | 30 weeks |
| | Isotopic Analysis for I-131, I-133, I-135 | Once per 14 days(3) | 20 days |
| 2. Boric Acid Tank | Boron Concentration, Chlorides | Weekly | 10 days |
| 3. Spray Additive Tank | NaOH Concentration | Monthly | 45 days |
| 4. Accumulators | Boron Concentration | Monthly | 45 days |
| SR 3.5.4.3 5. Refueling Water Storage Tank | Boron Concentration ~~pH, Chlorides~~ | Monthly 31 days | 45 days M.3 |
| SEE Relocated | Gross Activity | Quarterly | 16 weeks |
| 6. Secondary Coolant | I-131 Equivalent (Isotopic Analysis) | Monthly | 45 days |
| | Gross Activity | 3 times per 7 days | 3 days |
| 7. Component Cooling Water | Gross Activity, Corrosion Inhibitor and pH | Monthly | 45 days |
| 8. Spent Fuel Pool (when fuel stored) | Gross Activity Boron Concentration, Chlorides | Monthly | 45 days |

Amendment No. 139

ITS 3.5.4

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.5.4:
# "Refueling Water Storage Tank (RWST)"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.5.4 - Refueling Water Storage Tank (RWST)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety because neither are required by 10 CFR 50.36, and neither define nor impose any specific requirements.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.3.A.1 defines the Applicability and establishes requirements for the Refueling Water Storage Tank (RWST) whenever Tavg is greater than 200°F. CTS 3.3.A.3 defines the Applicability and establishes identical requirements for the RWST whenever Tavg is greater than 350°F. ITS LCO 3.5.4 maintains the same Applicability for the RWST as Modes 1, 2, 3 and 4. This is an administrative change with no impact on safety.

MORE RESTRICTIVE

M.1 CTS 3.3 and CTS 4.1 do not establish any acceptance criteria or require periodic verification of RWST water temperature. ITS SR 3.5.4.1 is added to establish acceptance criteria for the minimum and/or maximum RWST water temperature consistent with the limiting assumptions in any accident analysis and to require verification every 24 hours that minimum and/or maximum RWST water temperature is met. ITS SR 3.5.4.1 is modified by a note that eliminates the requirement to perform the surveillance when ambient air temperatures are within the RWST operating limits because RWST temperature should not exceed the limits when operating under these conditions.

In conjunction with this change, ITS LCO 3.5.4, Condition A and Required Action A.1, is added to require restoration of RWST temperature to within required limits within 8 hours or initiate reactor shutdown whenever ITS SR 3.5.4.1 is not met.

This change is needed to assure that the minimum and/or maximum RWST water temperature is maintained within the limits assumed in the accident analysis. This change is acceptable because it does not introduce any operation that is un-analyzed while requiring periodic verification that analysis assumptions regarding RWST water temperatures are satisfied. The 24 hour SR Frequency is sufficient to identify a temperature change that would approach either limit because the tank volume is large causing temperature change to be slow and because the tank is equipped with temperature control equipment. Therefore, this change has no significant adverse impact on safety.

M.2 CTS 3.3.A.5 establishes the Actions required if the ECCS systems (Refueling Water Storage Tank, Accumulators, Safety Injection Pumps, Residual Heat Removal Pumps, Recirculation Pumps) are not restored to meet CTS requirements within specified completion times when above 350 °F (Mode 3) (See ITS 3.5.4, DOC L.1). CTS 3.3.A.5.a specifies that, if the reactor is critical when requirements are not met, then the reactor shall be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within the following 24 hours. However, if the reactor is subcritical when requirements are not met, CTS 3.3.A.5.b requires only that reactor coolant system temperature and pressure not be increased more than 25°F and 100 psi, respectively, over existing values with the requirement to proceed to cold shutdown (Mode 5) deferred by another 48 hours.

Under the same conditions, ITS 3.5.4, Required Actions C.1 and C.2, require that the reactor be in Mode 3 in 6 hours (See ITS 3.5.1, DOC L.1) and Mode 5 in 36 hours regardless of the status of the unit when the Condition is identified. The allowance provided in CTS 3.3.A.5.b is deleted.

This change is needed to eliminate the ambiguity created by CTS 3.3.A.5.b when performing a reactor shutdown and cooldown required by CTS 3.3.A.5 and to ensure that the plant is placed outside the LCO Applicability promptly when the LCO requirements are not met. This change is acceptable because placing the plant outside the LCO Applicability when LCO requirements are not met is conservative and there is no change in the CTS 3.3.A.5 requirement. This change has no significant adverse impact on safety.

M.3 CTS Table 4.1-2, Item 5, specifies that the Frequency for verification of RWST boron concentration is 31 days and that the maximum time between accumulator boron concentration verification is 45 days.

ITS SR 3.5.4.3 maintains the requirement to verify RWST boron concentration every 31 days; however, ITS SR 3.0.2 limits any extension to the 31 day SR interval to 25% (approximately 39 days).

This change is needed to establish consistent allowances for extending SR Frequencies consistent with ITS SR 3.0.2. This change is acceptable because it does not introduce any operation that is un-analyzed while requiring more timely verification that analysis assumptions regarding the Operability of RWST boron concentration are satisfied. Therefore, this change has no significant adverse impact on safety.

M.4 CTS 3.3.A.1.a and CTS 3.3.A.3.a establish the requirement that RWST level is greater than 35.4 feet at all times the reactor is above cold shutdown; however CTS 3.3 and CTS 4.1 do not require periodic verification that RWST level is within required limits.

ITS SR 3.5.4.2 maintains the CTS acceptance criteria that RWST level is greater than 35.4 feet and includes the new requirement to verify every 7 days that RWST level is within required limits.

This change is needed to assure that the minimum and/or maximum RWST water level is maintained within the limits assumed in the accident analysis. This change is acceptable because it does not introduce any operation that is un-analyzed while requiring periodic verification that analysis assumptions regarding RWST water level are satisfied. The SR Frequency of 7 days is acceptable because RWST volume is normally stable and the tank is equipped with level indication and alarms. Therefore, this change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.3.A.2 requires the reactor be in cold shutdown (Mode 5) within 21 hours (1 hour plus 20 hours) if RWST required volume or boron concentration limits are not met whenever Tavg is greater than 200°F. CTS 3.3.A.5.a requires the reactor be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within 28 hours (4 hour plus 24 hours) if RWST required volume or boron concentration limits are not met whenever Tavg is greater than 350°F and the reactor is critical. CTS 3.3.A.5.b requires the that reactor coolant system pressure and temperature not be increased and a plant cooldown initiated within 48 hours if RWST required volume or boron concentration limits are not met whenever Tavg is greater than 350°F and the reactor is not critical (See ITS 3.5.4, DOC M.2).

Under the same conditions, ITS 3.5.4, Required Action C.1 and C.2, require the reactor be in Mode 3 in 6 hours and Mode 5 in 36 hours. This change is needed and is acceptable because, based on operating experience, Mode 3 in 6 hours and Mode 5 in 36 hours is a reasonable time to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. ITS 3.5.4, Required Action C.1 and C.2, still requires that the plant be promptly placed outside the LCO Applicability when requirements are not met. Therefore, this change has no significant adverse impact on safety.

L.2 CTS 3.3.A.2 and CTS 3.3.A.5.a require an RWST to be restored within one hour if it is inoperable because either level or boron concentration are not within required limits.

Under the same conditions, ITS 3.5.4, Required Action B.1, still requires that an RWST to be restored within one hour if it is inoperable because level is not within required limits; however, ITS 3.5.4, Required Action A.1, allows 8 hours to restore an RWST that is inoperable because boron concentration is not within limits. This change is needed because it provides a reasonable time to restore boron concentration to within required limits and avoids a plant shutdown for a parameter that can be corrected without reactor shutdown. This change is acceptable because the required volume of the RWST is still available for injection and boron concentration would be expected to be outside of required limits by only a small amount because boron concentration changes very slowly relative to the required Frequency of the SR that verifies boron concentration. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

None

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.5.4:
## "Refueling Water Storage Tank (RWST)"

## PART 4:

## No Significant Hazards Considerations
## for
## Changes between CTS and ITS
## that are
## Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.5.4 - Refueling Water Storage Tank (RWST)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS 3.5.4 extends the time allowed to complete a reactor shutdown to be in Mode 3 in 6 hours and Mode 5 in 36 hours if RWST required volume or boron concentration limits.

   This change will not result in a significant increase in the probability of an accident previously evaluated because the time required to perform a reactor shutdown when requirements are not met has no affect on the initiators of any analyzed events.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because , based on operating experience, Mode 3 in 6 hours and Mode 5 in 36 hours is a reasonable time to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. ITS 3.5.4, Required Actions C.1 and C.2, still require that the plant is promptly placed outside the LCO Applicability when requirements are not met.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way the RWST is operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because , based on operating experience, Mode 3 in 6 hours and Mode 5 in 36 hours is a reasonable time to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. ITS 3.5.4, Required Actions C.1 and C.2, still require that the plant is promptly placed outside the LCO Applicability when requirements are not met.

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LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   ITS 3.5.4 still requires that an RWST to be restored within one hour if it is inoperable because level is not within required limits; however, ITS 3.5.4, Required Action A.1, changes the CTS 3.3.A.2 and CTS 3.3.A.5.a and allows 8 hours to restore an RWST that is inoperable because boron concentration is not within limits before reactor shutdown is required.

   This change will not result in a significant increase in the probability of an accident previously evaluated because RWST boron concentration has no affect on the initiators of any analyzed events.

   This change will not result in a significant increase in the consequences of an accident previously evaluated because it provides a reasonable time to restore boron concentration to within required limits and avoids a plant shutdown for a parameter that can be corrected without reactor shutdown. This change is acceptable because the required volume of the RWST is still available for injection and boron concentration would be expected to be outside of required limits by only a small amount because boron concentration changes very slowly relative to the required Frequency of the SR that verifies boron concentration.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed changes will not involve any physical changes to plant systems, structures, or components (SSC). The changes in normal Plant operation are consistent with the current safety analysis assumptions because there is no change in the way the RWST is operated. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because it provides a reasonable time to restore boron concentration to within required limits and avoids a plant shutdown for a parameter that can be corrected without reactor shutdown. This change is acceptable because the required volume of the RWST is still available for injection and boron concentration would be expected to be outside of required limits by only a small amount because boron concentration changes very slowly relative to the required Frequency of the SR that verifies boron concentration.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.5.4:
# "Refueling Water Storage Tank (RWST)"

## PART 5:

## NUREG-1431
## Annotated to show differences between
## NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.5.4

This ITS Specification is based on NUREG-1431 Specification No. **3.5.4**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| N/A | N/A | NO GENERIC CHANGES ARE POSTED AGAINST THIS SPECIFICATION. | Not Applicable | Not Applicable | N/A |

## 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 3.5.4 Refueling Water Storage Tank (RWST)

<3.3.A.1.a>
<3.3.A.3.a>

LCO 3.5.4 The RWST shall be OPERABLE.

<3.3.A.1>
<3.3.A.3>
<DOC A.3>

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| | CONDITION | REQUIRED ACTION | COMPLETION TIME | |
|---|---|---|---|---|
| <3.3.A.2> <3.3.A.5> <DOC L.2> <DOC M.1> | A. RWST boron concentration not within limits of SR 3.5.4.3. <br> OR <br> RWST borated water temperature not within limits of SR 3.5.4.1. | A.1 Restore RWST to OPERABLE status. | 8 hours | PA.1 |
| <3.3.A.2> <3.3.A.5> | B. RWST inoperable for reasons other than Condition A. | B.1 Restore RWST to OPERABLE status. | 1 hour | |
| <DOC L.1> <DOC M.2> | C. Required Action and associated Completion Time not met. | C.1 Be in MODE 3. <br> AND <br> C.2 Be in MODE 5. | 6 hours <br><br> 36 hours | |

SURVEILLANCE REQUIREMENTS

| | | SURVEILLANCE | FREQUENCY | |
|---|---|---|---|---|
| <DOC M.1> | SR 3.5.4.1 | -------------------NOTE------------------- (remains) Only required to be performed when ambient air temperature is < [35] (40) °F or > [100] (100) °F. ----------------------------------------------- Verify RWST borated water temperature is ≥ [35] (40) °F and ≤ [100] (110) °F. | 24 hours (for 24 hours) | DB.2 |
| <DOC M.4> <3.3.A.1.a> <3.3.A.3.a> | SR 3.5.4.2 | Verify RWST borated water [volume] (level) is ≥ ~~[466,200 gallons (-%)]~~ (35.4 feet). | 7 days | |
| <T4.1-2,#5> <3.3.A.1.a> <3.3.A.3.a> | SR 3.5.4.3 | Verify RWST boron concentration is ≥ ~~[2000]~~ (2400) ppm and ≤ ~~[2200]~~ (2600) ppm. | ~~7~~ (31) days | |

## B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool Cavity during refueling, and to the ECCS and the Containment Spray System during to the ECCS to fill accumulators accident conditions.

The RWST supplies ~~both trains of~~ the ECCS and the Containment Spray System through separate~~, redundant~~ supply headers during the injection phase of a loss of coolant accident (LOCA) ~~recovery~~. ~~A~~ motor operated isolation valves ~~is~~ are provided ~~in each header~~ to isolate the RWST from the ECCS subsystems once the system has been transferred to the recirculation mode. ~~The recirculation mode is entered when pump suction is transferred to the containment sump following receipt of the RWST—Low Low (Level 1) signal.~~ Insert: B 3.5-25-01 Use of a single RWST to supply ~~both~~ All of the injection trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

~~The switchover from normal operation to the injection phase of ECCS operation requires changing centrifugal charging pump suction from the CVCS volume control tank (VCT) to the RWST through the use of isolation valves. Each set of isolation valves is interlocked so that the VCT isolation valves will begin to close once the RWST isolation valves are fully open. Since the VCT is under pressure, the preferred pump suction will be from the VCT until the tank is isolated. This will result in a delay in obtaining the RWST borated water. The effects of this delay are discussed in the Applicable Safety Analyses section of these Bases.~~

During normal operation in MODES 1, 2, and 3, the high head safety injection (SI) HHSI and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

The ECCS and Containment Spray System pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

(continued)

INSERT: B 3.5-25-01:

DB.1

The switchover to the cold leg recirculation phase is manually initiated when the RWST level has reached the low-low alarm setpoint and sufficient coolant inventory to support pump operation in recirculation mode is verified to be in the containment.

Recirculation sump or the

BASES

BACKGROUND (continued)

~~When the suction for the ECCS and Containment Spray System pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.~~

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA. or MSLB

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment. due to improper pH in the sumps

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS—Operating"; B 3.5.3, "ECCS—Shutdown"; and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

Containment Fan Coolers

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. ~~The importance of its value is small for units with a boron injection tank (BIT) with a high boron concentration. For units with no BIT or reduced BIT boron requirements, the minimum boron concentration limit is an important assumption in ensuring the required shutdown capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.~~

Move to Page B3.5-28

~~The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as [27] seconds, with offsite power available, or [37] seconds without offsite power. This response time includes [2] seconds for electronics delay, a [15] second stroke time for the RWST valves, and a [10] second stroke time for the VCT valves. Plants with a BIT need not be concerned with the delay since the BIT will supply highly borated water prior to RWST switchover, provided the BIT is between the pumps and the core.~~

195,800

For a large break LOCA analysis, the minimum water volume limit of ~~[466,200]~~ gallons and the lower boron concentration limit of ~~[2000]~~ ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

2300

Insert: B3.5-27-01

The upper limit on boron concentration of ~~[2200]~~ ppm is used to determine the maximum allowable time to switch to hot leg

2600

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.5.4 - Refueling Water Storage Tank (RWST)

INSERT: B 3.5-27-01:

The RWST level required by Technical specifications includes allowances for instrument accuracy, the unuseable volume in the RWST, and the maximum volume expected to remain in the RWST when the plant is switched from the injection to recirculation modes of operation.

BASES

APPLICABLE SAFETY ANALYSES (continued)

recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 40 35°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of 110 100°F is used ~~in the small break LOCA analysis~~ and containment ~~OPERABILITY~~ in the LOCA Containment integrity analysis. Exceeding this temperature will result in ~~a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA and~~ higher containment pressures due to reduced containment spray cooling capacity. Insert from Page B3.5-27 For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment Recirculation sump and sump to support ECCS ~~and Containment Spray System~~ pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS

(continued)

BASES

4

**APPLICABILITY (continued)**

Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

5

**ACTIONS**

A.1

of SR 3.5.4.3 and SR 3.5.4.1, respectively

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.

unless

not

for more than 24 hours

SR 3.5.4.2

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS ~~and Containment Spray~~ System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

31

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron

31

level

(continued)

BASES

| | |
|---|---|
| SURVEILLANCE REQUIREMENTS | SR 3.5.4.3 (continued)<br><br>concentration is appropriate and has been shown to be acceptable through operating experience. |
| REFERENCES | 1. FSAR, Chapter [6] and Chapter ~~[15]~~ 14. |

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.5.4:
# "Refueling Water Storage Tank (RWST)"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

## JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
## ITS SECTION 3.5.4 - Refueling Water Storage Tank (RWST)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical errors or made a minor editorial improvements to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, these changes are not significant or generic deviations from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point
3

Docket # 50-286
Accession # 9812150197
Date 12/11/98 of Ltr
Regulatory Docket File

# Improved

# Technical Specifications

# Conversion Submittal

Volume 10

New York Power
Authority

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.1:
## "Containment"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.6 CONTAINMENT SYSTEMS

### 3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Containment inoperable. | A.1 Restore containment to OPERABLE status. | 1 hour |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 5. | 6 hours<br><br><br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.1.1 | Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program. | -----NOTE------<br>SR 3.0.2 is not applicable<br>--------------<br><br>In accordance with the Containment Leakage Rate Testing Program |

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete reactor building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA), in particular, a Main Steam Line Break (MSLB) inside containment or a Loss of Coolant Accident (LOCA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The concrete reactor building is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B, (Ref. 1), as established in Specification 5.5.15, Containment Leakage Rate Testing Program.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic containment isolation system, or

BASES

BACKGROUND (Continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";

c. The equipment hatch is properly closed; and

d. The Isolation Valve Seal Water (IVSW) system is OPERABLE, except as provided in LCO 3.6.9.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day assuming the proper functioning of the Isolation Valve Seal Water System but without benefit of the Weld Channel and Penetration Pressurization System (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 1), as $L_a$: the maximum allowable containment leakage rate at the calculated peak containment internal pressure ($P_a$) resulting from the limiting DBA. The allowable leakage rate represented by $L_a$ forms the basis for the acceptance criteria imposed on all containment leakage rate testing. $L_a$ is assumed

BASES

APPLICABLE SAFETY ANALYSES (continued)

to be 0.1% of containment air weight per day in the safety analysis at $P_a$ which is specified in Specification 5.5.15, Containment Leakage Rate Testing Program.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36.

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test in accordance with requirements in Specification 5.5.15, Containment Leakage Rate Testing Program. At this time, the applicable leakage limits specified in the Containment Leakage Rate Testing Program must be met.

Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to less than the leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air locks (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the acceptance criteria of Appendix J.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The

BASES

APPLICABILITY (continued)

requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in LCO 3.6.2 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1 (continued)

startup after performing a required leakage test is required to be $\leq$ 0.6 $L_a$ for combined Type B and C leakage following an outage or shutdown that included Type B and C testing only, and $\leq$ 0.75 $L_a$ for overall Type A leakage following an outage or shutdown that included Type A testing. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq$ 1.0 $L_a$. At $\leq$ 1.0 $L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR, Chapter 14.
3. FSAR, Section 6.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.1:
## "Containment"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 1-4 | 34 TSCR 97-070 | 34 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| 3.6-1 | 86 TSCR 97-070 | 86 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| 3.6-3 | 8-30-95 TSCR 97-070 | 8-30-95 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| 4.4-1 | 174 | 174 | No TSCRs | No TSCRs for this Page | N/A |
| 4.4-2 | 174 | 174 | No TSCRs | No TSCRs for this Page | N/A |
| 4.4-5 | 174 | 174 | No TSCRs | No TSCRs for this Page | N/A |
| 4.4-7 | 174 TSCR 98-043 | 174 TSCR 98-043 | IPN 98-043 | Instrument Channel Surveillance Intervals Extended to 24 Months | Incorporated |
| 4.4-10 | 174 TSCR 98-043 | 174 TSCR 98-043 | IPN 98-043 | Instrument Channel Surveillance Intervals Extended to 24 Months | Incorporated |

1.9.2 Instrument Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating actions.

1.9.3 Instrument Channel Calibration

SEE ITS 1.0

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the channel functional test.

1.9.4 Logic Channel Functional Test

The operation of relays or switch contacts, in all the combinations required, to produce the required output.

~~1.10~~ CONTAINMENT INTEGRITY

LCO 3.6.1 Containment integrity is defined to exist when: A.3

SEE ITS 3.6.3

~~1.10.1~~ Non-automatic containment isolation valves (Table 3.6-1) are closed or may be opened under administrative control and only as long as necessary to perform their intended function.

~~1.10.2~~ Blind flanges, that provide an isolation function which are shown in FSAR drawings, are maintained installed.

~~1.10.3~~ Any test connection, vent or drain valve that is located within the isolation boundary and is required to perform an isolation function is closed and capped (threaded) or blind flanged as shown in FSAR drawings.

~~1.10.4~~ ~~The equipment door is properly closed.~~ LA.1

SEE ITS 3.6.2

~~1.10.5~~ Both doors in each personnel air lock are properly closed unless being used for entry, egress or maintenance, at which time at least one air lock door shall be closed.

SEE ITS 3.6.3

~~1.10.6~~ All automatic containment isolation valves are either operable or in the closed position, or isolated by a closed manual valve or flange that meets the same design criteria as the isolation valve.



Amendment No. 34,

TSCR 97-070

3.6 CONTAINMENT SYSTEM

~~Applicability~~

~~Applies to the integrity of reactor containment.~~

~~Objective~~

~~To define the operating status of the reactor containment for plant operation.~~

~~Specification~~

A.2

A.3

A. Containment Integrity

shall be Operable

Mode 1, 2, 3 & 4 — A.4

LCO 3.6.1 ~~1.~~ Containment ~~integrity (as defined in 1.10) shall not be violated~~ unless the reactor is in the cold shutdown condition. Those valves to be opened continuously or intermittently are under administrative control and are open only as long as necessary to perform their intended function.

SEE ITS 3.6.3

SEE ITS 3.9.1 2. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is sufficient to maintain the shutdown margin equal to or greater than the requirements of specification 3.8.D.

Req Act A.1 ~~3.~~ If the containment integrity requirements are not met when the reactor is above cold shutdown containment integrity shall be restored within one hour or the reactor shall be in the ~~hot shutdown~~ Mode 3 condition within six hours and in cold shutdown Mode 5 condition within the next 30 hours.

Req. Act B.1 B.2

A.4

B. Internal Pressure

SEE ITS 3.6.4

If the internal pressure exceeds 2.5 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor shutdown.



Amendment No. ~~14~~, ~~66~~

TSCR 97-070

Regarding internal pressure limitations, the containment design pressure of 47 psig would not be exceeded for a major loss-of-coolant accident or for a main steam line break accident.[1] The loss-of-coolant accident event bounds the main steam line break accident from the containment peak pressures standpoint. The initial pressure condition used in the containment analysis was 2.5 psig.[1] The containment can withstand an internal vacuum of 3 psig.[2] The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

The requirement of a 50°F minimum containment ambient temperature is to assure that the minimum service metal temperature of the containment liner is well above the NDT + 30°F criterion for the liner material. [3]

Limiting maximum containment ambient temperature will ensure that the peak accident containment pressure does not exceed the design limit of 47 psig during steamline break or loss of coolant accidents. Environmentally and seismically qualified RTDs mounted on the crane wall above the containment fan cooler units inlet are normally used for measuring containment ambient temperature. Portable temperature sensing equipment may also be used, provided the criteria of 3.6.C.3 are met.

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function. [4] During periods of normal plant operations requiring containment integrity, some of the valves on this Table will be open either continuously or intermittently depending on requirements of the particular protection, safeguards or essential service systems. Those valves to be opened are under administrative control and are open only as long as necessary to perform their intended function. Some of the valves listed in Table 3.6-1 are closed during the post accident period in accordance with plant procedures and consistent with requirements of the related protection, safeguards, or essential service systems.

The opening angle of the containment vent isolation valves is being limited as an analysis demonstrates valve operability against accident containment pressures provided the valves are limited to a maximum opening angle of 60°. The containment purge supply and exhaust isolation valves are required to be closed during plant operation above cold shutdown.

REFERENCES

(1) FSAR - Section 14.3.6
(2) FSAR - Appendix 5A, Section 3.1.8
(3) FSAR - Section 5.1.1.1
(4) FSAR - Section 5.2

4.4 CONTAINMENT TESTS

~~Applicability~~

~~Applies to containment leakage.~~

~~Objective~~

~~To verify that potential leakage from the containment is maintained within acceptable values.~~

~~Specification~~

A2

~~A~~ Integrated Leakage Rate

SR3.6.1.1 Perform required visual examinations and leakage rate testing, except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program.

~~B. DELETED~~ A.1

F. ~~DELETED~~

G. ~~DELETED~~

H. ~~DELETED~~

A.1

**Basis**

The containment is designed for a pressure of 47 psig. [1] While the reactor is operating, the internal environment of the containment will be air at essentially atmospheric pressure and an average maximum temperature of approximately 130°F. The Design Basis Accidents (DBA) that represent a challenge to the containment structure are a Loss of Coolant Accident (LOCA) and a Main Steam Line Break (MSLB). The limiting calculated peak containment pressure of 42.40 psig is a result of the MSLB [7], which is less than the stated design pressure of 47 psig. In addition, DBA analyses demonstrate that the calculated peak containment temperature will remain less than the Equipment Qualification (EQ) envelope temperature of 290 degrees F.

The containment structure is designed to contain, within established leakage limits, radioactive material that may be released from the reactor core following a DBA. The containment was designed with an allowable leakage rate of 0.1 weight percent of containment air per day. This leakage rate, used to evaluate offsite doses resulting from DBAs is defined in 10CFR 50 Appendix B as $L_a$; the maximum allowable containment leakage rate at the calculated peak containment internal pressure ($P_a$) resulting from the limiting DBA. The allowable leakage rate represented by $L_a$ forms the basis for the acceptance criteria imposed on all containment leakage rate testing performed in accordance with the program required by Technical Specification 6.14. The minimum test pressure for this program, based on the current value of $P_a$ is 42.40 psig. Analyses which established the previous minimum test pressure of 42.42 psig were performed to support an increase of the ultimate heat sink temperature. [4] The conclusions of that analysis regarding heat sink temperature, as incorporated by Technical Specification Amendment 98, remain valid.

Prior to initial operation, the containment was strength-tested at 54 psig and was leak-tested. The acceptance criterion for this pre-operational leakage rate test was established as 0.075 W/o (.75 $L_a$) per 24 hours at 40.6 psig and 263°F, which were the peak accident pressure and temperature conditions at that time. This leakage rate is consistent with the construction of the containment, [2] which is equipped with a Weld Channel and Penetration Pressurization System for continuously pressurizing the containment penetrations and the channels over certain containment liner welds. These channels were independently leak-tested during construction.

The safety analysis has been performed on the basis of a leakage rate of 0.10 W/o per day for 24 hours. With this leakage rate and with minimum containment engineered safeguards operating, the public exposure would be well below 10CFR100 values in the event of the design basis accident. [3]

Maintaining the containment operable requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in surveillance requirement 4.4.D does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage

~~The maximum permissible inleakage rate from the containment isolation valves sealed with service water for the full 12-month period of post accident recirculation without flooding the internal recirculation pumps is 0.36 gpm per fan cooler.~~


~~REFERENCES~~

~~(1) FSAR - Section 5~~
~~(2) FSAR - Section 5.1.7~~
~~(3) FSAR - 14.3.5~~
~~(4) WCAP - 12269 Rev. 1, "Containment Margin Improvement Analysis for IP-3 Unit 3"~~
~~(5) FSAR - Section 6.6~~
~~(6) FSAR - Section 6.5~~
~~(7) Nuclear Safety Evaluation 98-3-013-MULT, "Integrated Safety Evaluation of 24-Month Cycle Instrument Channel Uncertainties," Revision 0, dated March 3, 1998.~~
~~(8) SECL-96-103, Indian Point Unit 3 Safety Evaluation of 24-Month Fuel Cycle Phase I Instrument Channel Uncertainties, June 1996~~
~~(9) Indian Point 3 Safety Evaluation Report, Supplement 2, December 1975.~~
~~(10) NRC Safety Evaluation Related to Amendment 129 to Operating License DPR-64.~~


A.1

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.1:
## "Containment"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.6.1 - Containment

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.6.A.1 specifies that containment integrity (as defined in CTS 1.10) shall not be violated; and, CTS 1.10 specifies that containment integrity exists when isolation valves, air locks, and the equipment hatch are set as necessary to maintain the leak tightness of the containment. Additionally, CTS 4.4.A requires that visual examinations and leakage rates of the containment shall be in accordance with the Containment Leakage Rate Testing Program.

The ITS maintains these requirements by dividing the containment Operability requirements into four separate LCOs: ITS 3.6.1 which requires that the containment is Operable; ITS 3.6.2 which requires that the containment air locks are Operable; ITS 3.6.3 which requires that each containment isolation valve is Operable; and, ITS LCO 3.6.9 which requires that IVSW is Operable. In conjunction with this change, the CTS definition of Containment Integrity is deleted because it contains information that is more appropriately contained in the LCOs (and SRs) which establish the requirements for containment integrity and the Bases associated with these LCOs and SRs. This reorganization ensures that appropriate LCOs are recognized for any Condition and that appropriate Required Actions are implemented.

This change is needed to improve clarity and ensure requirements are fully understood and consistently applied. This reorganization of requirements is an administrative change with no impact on safety except for the specific changes identified and justified in the discussion of changes for each LCO addressing containment issues.

A.4 CTS 3.6.A.1 and CTS 3.6.A.3 specify the Applicability for containment Integrity as whenever the reactor is above cold shutdown (i.e., Modes 1, 2, 3 and 4). ITS 3.6.1 maintains this Applicability by requiring that Containment is Operable in Modes 1, 2, 3 and 4. This is an administrative change with no impact on safety because there is no change to the CTS Applicability.

MORE RESTRICTIVE

None

LESS RESTRICTIVE

None

REMOVED DETAIL

LA.1 CTS 1.10.2 specifies that the equipment door (hatch) must be properly closed as a condition of containment integrity. LCO 3.6.1 and associated SRs do not specifically address the status of the containment equipment hatch as a requirement for containment Operability; however, the Bases for LCO 3.6.1 specify that the equipment hatch must be closed. Moving this detail of containment Operability to the Bases is acceptable because SR 3.6.1.1, periodic leakage rate testing, includes a specific requirement for visual examination of the containment which will ensure that the equipment hatch is properly closed (i.e., in accordance with design drawings). Additionally, SR 3.6.1.1 acceptance criteria must be assumed not met if the equipment hatch is not properly installed at any time between performances of the SR. Therefore, the requirement to have the equipment hatch properly installed is not changed and is enforced indirectly by SR 3.6.1.1 and the description of requirements for Operability in the ITS Bases. Therefore, this design information can be adequately defined and controlled in the ITS Bases which require change control in accordance with ITS 5.5.13, Bases Control Program. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the requirement to maintain the equipment hatch closed as a condition of containment Operability. This change is a less restrictive administrative change with no impact on safety.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.6.1:
# "Containment"

## PART 4:

## No Significant Hazards Considerations
## for
## Changes between CTS and ITS
## that are
## Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.6.1 - Containment

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

There are no less restrictive changes for the adoption of this ITS.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.6.1:
# "Containment"

## PART 5:

## NUREG-1431
## Annotated to show differences between NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.6.1

This ITS Specification is based on NUREG-1431 Specification No. **3.6.1**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| CEOG-112 | **196 R0** | REVISE ISOLATION DEVICES TO INCLUDE ASME/ANSI EQUIVALENT METHODS | Rejected by NRC | Not Incorporated. OI to evaluate if benefit. | **N/A** |
| WOG-042 | **052** | IMPLEMENT 10 CFR 50, APPENDIX J, OPTION B | TSTF to Rewrite | Plant specific adoption of Option B incorporated. | **N/A** |

## 3.6 CONTAINMENT SYSTEMS

### 3.6.1 Containment ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~

<1.10>
<3.6.A.1>
<DOC A.3>

LCO 3.6.1 Containment shall be OPERABLE.

<3.6.A.1>
<DOC A.4>

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <3.6.A.3> | A. Containment inoperable. | A.1 Restore containment to OPERABLE status. | 1 hour |
| <3.6.A.3> | B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3. | 6 hours |
| | | <u>AND</u> | |
| | | B.2 Be in MODE 5. | 36 hours |

<CTS>

<4.4.A>

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.1.1 | Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~ the Containment Leakage Rate Testing Program<br><br>~~The leakage rate acceptance criterion is $\le$ 1.0 $L_a$. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are < 0.6 $L_a$ for the Type B and Type C tests, and < 0.75 $L_a$ for the Type A test.~~ | -----NOTE------<br>SR 3.0.2 is not applicable<br>--------------<br><br>In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ |
| ~~SR 3.6.1.2~~ | ~~Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.~~ | ~~In accordance with the Containment Tendon Surveillance Program~~ |

CLB.1

## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.1 Containment ~~(Atmospheric)~~

BASES

BACKGROUND

The containment consists of the concrete reactor building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

Insert: B3.6-6-01

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a ~~shallow~~ dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

~~For containments with ungrouted tendons, the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed utilizing a three way post tensioning system.~~

The concrete reactor building is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B, (Ref. 1), as ~~modified by approved exemptions~~. T.1

Insert: B3.6-6-02

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

1. capable of being closed by an OPERABLE automatic containment isolation system, or

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.6.1 - Containment

**INSERT: B 3.6-6-01**

, in particular, a Main Steam Line Break (MSLB) inside containment or a Loss of Coolant Accident (LOCA).

**INSERT: B 3.6-6-02**

established in Specification 5.5.15, Containment Leakage Rate Testing Program.

BASES

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";

c. The ~~All~~ equipment hatch~~es~~ ~~are~~ is properly closed; and

Insert: B 3.6-7-01

d. ~~The pressurized sealing mechanism associated with a penetration is OPERABLE, except as provided in LCO 3.6.[ ].~~

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), and a steam line break~~, and a rod ejection accident (REA)~~ (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA ~~or REA~~. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~[0.1]%~~ of containment air weight per day (Ref. 3).

Insert: B 3.6-7-02

This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 1), as $L_a$: the maximum allowable containment leakage rate at the calculated peak containment internal pressure ($P_a$) resulting from the limiting DBA. The allowable leakage rate represented by $L_a$ forms the basis for the acceptance criteria imposed on all containment leakage rate testing. $L_a$ is assumed to be ~~[0.1]%~~ of containment air weight per day in the safety analysis at $P_a$ ~~= [44.1] psig~~ ~~(Ref. 3)~~.

Insert: B 3.6-7-03

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.6.1 - Containment

INSERT: B 3.6-7-01

DB.1

d. The Isolation Valve Seal Water (IVSW) system is OPERABLE, except as provided in LCO 3.6.9.

INSERT: B 3.6-7-02

DB.1

assuming the proper functioning of the Isolation Valve Seal Water System but without benefit of the Weld Channel and Penetration Pressurization System

INSERT: B 3.6-7-03

which is specified in Specification 5.5.15, Containment Leakage Rate Testing Program.

BASES

**APPLICABLE SAFETY ANALYSES (continued)**

The containment satisfies Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36

**LCO**

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$~~, except prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test. At this time, the combined Type B and C leakage must be $< 0.6\ L_a$, and the overall Type A leakage must be $< 0.75\ L_a$~~.

Insert: B 3.6-8-01

Insert: B 3.6-8-02

(T.1)

Compliance with this LCO will ensure a containment configuration, including equipment hatch~~es~~, that is structurally sound and that will limit leakage to ~~those~~ leakage rates assumed in the safety analysis.

the

less than the

Individual Δ leakage rates specified for the containment air lock (LCO 3.6.2) ~~[and purge valves with resilient seals (LCO 3.6.3)]~~ are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the acceptance criteria of Appendix J.

**APPLICABILITY**

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.~~4~~ 3, "Containment Penetrations."

**ACTIONS**

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.6.1 - Containment

INSERT: B 3.6-8-01

in accordance with requirements in Specification 5.5.15, Containment Leakage Rate Testing Program.

INSERT: B 3.6-8-02

the applicable leakage limits specified in the Containment Leakage Rate Testing Program must be met.

BASES

ACTIONS

A.1 (continued)

also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Insert: B 3.6-9-01

Insert: B3.6-9-02

Insert: B3.6-9-03

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of ~~10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.~~ Failure to meet air lock ~~[and purge valve with resilient seal]~~ leakage limits specified in LCO 3.6.2 ~~[and LCO 3.6.3]~~ does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required 10 CFR 50 Appendix J leakage test is required to be < 0.6 $L_a$ for combined Type B and C leakage, and < 0.75 $L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 $L_a$. At ≤ 1.0 $L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by ~~Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~ These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis. ≤ T.1

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.6.1 - Containment

**INSERT: B 3.6-9-01**

the Containment Leakage Rate Testing Program

**INSERT: B 3.6-9-02**

following an outage or shutdown that included Type B and C testing only

**INSERT: B 3.6-9-03**

following an outage or shutdown that included Type A testing

BASES

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~SR 3.6.1.2~~

~~For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 4).~~

REFERENCES

1. 10 CFR 50, Appendix J. , Option B
2. FSAR, Chapter ~~15~~. 14
3. FSAR, Section ~~[6.2]~~. 6
4. ~~Regulatory Guide 1.35, Revision [1].~~

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.1: "Containment"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.6.1 - Containment

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 This change maintains IP3 current licensing basis related to the use of 10 CFR 50, Appendix J, Option B, for containment leak rate testing which was approved on June 17, 1997 as part of Amendment 174. This change is based on Generic Change TSTF-52 (WOG-42), Revision 0, which is currently being reviewed by the NRC.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates 10 CFR 50, Appendix J, Option B, for containment leak rate testing. It is based on Generic Change TSTF-52 (WOG-42), Revision 0, which incorporates 10 CFR 50, Appendix J, Option B, which is currently under review. Acceptance Criteria are based on the plant specific methods submitted to and approved by the NRC.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.2:
# "Containment Air Locks"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

# 3.6 CONTAINMENT SYSTEMS

## 3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

------------------------------------NOTES------------------------------------

1. Entry and exit is permissible to perform repairs on the affected air lock components.

2. Separate Condition entry is allowed for each air lock.

3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate.

-----------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more containment air locks with one containment air lock door inoperable. | ------------NOTES------------<br>1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered.<br><br>2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable.<br>----------------------------- | |
| | | (continued) |

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. (continued) | A.1 | Verify the OPERABLE door is closed in the affected air lock. | 1 hour |
| | AND | | |
| | A.2 | Lock the OPERABLE door closed in the affected air lock. | 24 hours |
| | AND | | |
| | A.3 | --------NOTE---------- Air lock doors in high radiation areas may be verified locked closed by administrative means. --------------------- | |
| | | Verify the OPERABLE door is locked closed in the affected air lock. | Once per 31 days |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. One or more containment air locks with containment air lock interlock mechanism inoperable. | -------------NOTES------------- 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. ------------------------------- | |
| | B.1 Verify an OPERABLE door is closed in the affected air lock. | 1 hour |
| | AND | |
| | B.2 Lock an OPERABLE door closed in the affected air lock. | 24 hours |
| | AND | |
| | B.3 ---------NOTE--------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------------------- Verify an OPERABLE door is locked closed in the affected air lock. | Once per 31 days |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| C. One or more containment air locks inoperable for reasons other than Condition A or B. | C.1 | Initiate action to evaluate overall containment leakage rate per LCO 3.6.1. | Immediately |
| | <u>AND</u> | | |
| | C.2 | Verify a door is closed in the affected air lock. | 1 hour |
| | <u>AND</u> | | |
| | C.3 | Restore air lock to OPERABLE status. | 24 hours |
| D. Required Action and associated Completion Time not met. | D.1 | Be in MODE 3. | 6 hours |
| | <u>AND</u> | | |
| | D.2 | Be in MODE 5. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.2.1 | ------------------NOTES------------------<br>1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.<br><br>2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1, the integrated leak rate test.<br>-----------------------------------------------<br><br>Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program. | In accordance with the Containment Leakage Rate Testing Program |
| SR 3.6.2.2 | Verify only one door in the air lock can be opened at a time. | 24 months |

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is a cylinder with a door at each end. One of the two air locks is designed as a part of the containment structure and the other is designed as an integral part of the containment equipment hatch but otherwise the two air locks function identically. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY.

Each air lock door and the equipment hatch is designed with double gasketed seals to permit pressurization between the gaskets. The double gasketed seals are normally continuously pressurized above accident pressure. Finally, to effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door) and local leakage rate testing capability is available to ensure containment integrity is being maintained.

The doors are interlocked to prevent simultaneous opening of the inner and outer door. This interlock is a requirement for OPERABILITY. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary.

Each personnel air lock is provided with limit switches on both doors that provide control room indication when an airlock door is not fully closed.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is

BASES

BACKGROUND (continued)

essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident. In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as $L_a$ = 0.1% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure following a DBA. The peak pressure following a DBA is specified in Specification 5.5.15, Containment Leakage Rate Testing Program. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36.

LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The

BASES

LCO (continued)

interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

Pressurization of air lock seals is not required for air lock OPERABILITY. However, 10 CFR 50, Appendix J, Section III.2.b(iii), specifies that air locks opened during periods when containment integrity is required must be tested within 3 days after being opened. However, for air lock doors having testable seals, testing the seals (i.e., verification that seals re-pressurize to the required pressure after an air lock door is closed) fulfills the 3-day test requirements. Therefore, the status of air lock seals has the potential to affect air lock OPERABILITY.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. When the inner door is inoperable, it is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be

BASES

ACTIONS (continued)

performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for

BASES

ACTIONS

A.1. A.2. and A.3 (continued)

locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

during the short time that the OPERABLE door is expected to be open.

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to

BASES

ACTIONS

C.1, C.2, and C.3 (continued)

OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time unless Condition C is exited in accordance with LCO 3.0.2 (i.e., one door OPERABLE). The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), in accordance with Specification 5.5.15, Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1 (continued)

initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is in accordance with Specification 5.5.15, Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria that is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit, this test is only required to be performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if

BASES
SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

the Surveillance were performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience. The Frequency is based on engineering judgment and is considered adequate given that the interlock is not normally challenged during the use of the airlock.

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REFERENCES

1. 10 CFR 50, Appendix J.

2. FSAR, Section 6.6.

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Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.2: "Containment Air Locks"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 1-4 | 34 TSCR 97-070 | 34 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| 3.6-1 | 86 TSCR 97-070 | 86 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| 3.6-3 | 8-30-95 TSCR 97-070 | 8-30-95 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| 4.4-3 | 174 | 174 | No TSCRs | No TSCRs for this Page | N/A |
| 4.4-8 | 174 TSCR 98-043 | 174 TSCR 98-043 | IPN 98-043 | Instrument Channel Surveillance Intervals Extended to 24 Months | Incorporated |

~~1.9.2~~ Instrument Channel Functional Test

SEE ITS 1.0

Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating actions.

~~1.9.3~~ Instrument Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the channel functional test.

~~1.9.4~~ Logic Channel Functional Test

The operation of relays or switch contacts, in all the combinations required, to produce the required output.

~~1.10~~ CONTAINMENT INTEGRITY

LCO 3.6.2

Containment integrity is defined to exist when: A.3

SEE ITS 3.6.3

~~1.10.1~~ Non-automatic containment isolation valves (Table 3.6-1) are closed or may be opened under administrative control and only as long as necessary to perform their intended function.

~~1.10~~.2 Blind flanges, that provide an isolation function which are shown in FSAR drawings, are maintained installed.

1.10.3 Any test connection, vent or drain valve that is located within the isolation boundary and is required to perform an isolation function is closed and capped (threaded) or blind flanged as shown in FSAR drawings.

SEE ITS 3.6.1 1.10.4 The equipment door is properly closed. A.3

LCO 3.6.2 ~~1.10.5~~ Both doors in each two personnel air lock are ~~properly closed~~ Operable ~~unless being used for entry, egress or maintenance, at which time at least one air lock door shall be closed~~. A.3

SEE ITS 3.6.3

1.10.6 All automatic containment isolation valves are either operable or in the closed position, or isolated by a closed manual valve or flange that meets the same design criteria as the isolation valve.



Amendment No. 34,

TSCR 97-070

LCO 3.6.2

# ITS 3.6.2

Add Condition B and Note 2 to Cond B Req Acts — M.2

3.6 CONTAINMENT SYSTEM

~~Applicability~~

~~Applies to the integrity of reactor containment.~~

~~Objective~~

~~To define the operating status of the reactor containment for plant operation.~~ — A.2

~~Specification~~

Operable — A.3

Two containment air locks

Mode 1, 2, 3 and 4 — A.4

A. Containment Integrity

LCO 3.6.2 ~~1.~~ Containment integrity ~~as defined in 1.10~~ shall ~~not~~ be ~~violated~~ ~~unless~~ the reactor is in the cold shutdown condition. Those valves to be opened continuously or intermittently are under administrative control and are open only as long as necessary to perform their intended function.

SEE ITS 3.6.3

2. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is sufficient to maintain the shutdown margin equal to or greater than the requirements of specification 3.8.D.

SEE ITS 3.9.1

LCO 3.6.2 ~~3.~~ Req Act A.1, B.1, C.2; Req Act D.1, D.2

If the containment integrity requirements are not met when the ~~reactor is above cold shutdown~~ containment integrity shall be restored within one hour or the reactor shall be in the hot shutdown condition within six hours and in cold shutdown condition within the next 30 hours. — A.4, A.5

Mode 5 Mode 3

B. Internal Pressure

SEE ITS 3.6.4

If the internal pressure exceeds 2.5 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor shutdown.

LCO 3.6.2 Add Actions Note 1 and Note 2 to Req Act A.1 — L.1

Add Actions Note 2 — A.6

Add Req Acts C.1, C.3 — L.2

Add Actions Note 3 — A.7

Add Req Act A.1, B.1, C.2 — A.5

3.6-1 Add Note 1 to Req Act A.1, B.1 — A.8

Add Req Act A.2, A.3 B.2, B.3 — M.1

Amendment No. ~~34~~, ~~88~~

TSCR 97-070

A-1

Regarding internal pressure limitations, the containment design pressure of 47 psig would not be exceeded for a major loss-of-coolant accident or for a main steam line break accident.[1] The loss-of-coolant accident event bounds the main steam line break accident from the containment peak pressures standpoint. The initial pressure condition used in the containment analysis was 2.5 psig.[1] The containment can withstand an internal vacuum of 3 psig.[2] The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

The requirement of a 50°F minimum containment ambient temperature is to assure that the minimum service metal temperature of the containment liner is well above the NDT + 30°F criterion for the liner material. [3]

Limiting maximum containment ambient temperature will ensure that the peak accident containment pressure does not exceed the design limit of 47 psig during steamline break or loss of coolant accidents. Environmentally and seismically qualified RTDs mounted on the crane wall above the containment fan cooler units inlet are normally used for measuring containment ambient temperature. Portable temperature sensing equipment may also be used, provided the criteria of 3.6.C.3 are met.

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function. [4] During periods of normal plant operations requiring containment integrity, some of the valves on this Table will be open either continuously or intermittently depending on requirements of the particular protection, safeguards or essential service systems. Those valves to be opened are under administrative control and are open only as long as necessary to perform their intended function. Some of the valves listed in Table 3.6-1 are closed during the post accident period in accordance with plant procedures and consistent with requirements of the related protection, safeguards, or essential service systems.

The opening angle of the containment vent isolation valves is being limited as an analysis demonstrates valve operability against accident containment pressures provided the valves are limited to a maximum opening angle of 60°. The containment purge supply and exhaust isolation valves are required to be closed during plant operation above cold shutdown.

REFERENCES

(1) FSAR - Section 14.3.6
(2) FSAR - Appendix 5A, Section 3.1.8
(3) FSAR - Section 5.1.1.1
(4) FSAR - Section 5.2



Amendment No. ~~62, 86, 98~~, ~~Revised by letter dated 8/30/95~~,

TSCR 97-070

C. Sensitive Leakage Rate

SEE RELOCATED CTS

Verify the leakage rate for the Containment Penetration and Weld Channel Pressurization System is ≤ 0.2 percent of the containment free volume per day when pressurized to ≥ 43 psig and the containment pressure is atmospheric. The testing shall be performed at intervals no greater than 3 years.

D. Air Lock Tests

SR 3.6.2.1

Perform required Containment Air Lock leak rate testing in accordance with the Containment Leakage Rate Testing Program.

Add SR 3.6.2.1, Note 1 — A.10

Add SR 3.6.2.1, Note 2 — A.9

Add SR 3.6.2.2 — M.3

prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test is required to be <0.6 $L_a$ for combined Type B and C leakage, and < 0.75 $L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 $L_a$. At ≤ 1.0 $L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. Surveillance requirement frequencies are as required by the Containment Leakage Rate Testing Program. Thus, Specification 1.12 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

A.1

The Weld Channel and Containment Penetration Pressurization System (WCCPPS)[5] is in service continuously to monitor leakage from potential leak paths such as the containment personnel lock seals and weld channels, containment penetrations, containment liner weld channels, double-gasketed seals and spaces between certain containment isolation valves and personnel door locks. A leak would be expected to build up slowly and would, therefore, be noted before design limits are exceeded. Remedial action can be taken before the limit is reached. The sensitive leakage rate test of the WCCPPS demonstrates that pressurized containment penetrations and liner inner weld seams are within a leakage acceptance criteria that will allow the air receivers and the standby source of gas pressure, nitrogen cylinders, to provide a 24 hour supply of gas to the system. The WCCPPS is not credited for limiting containment isolation valve leakage and the sensitivity test is not used for demonstrating compliance with containment isolation valve leakage criteria. The frequency of the sensitive leakage test reflects an extension of 25 percent from the 24 month refueling cycle and, therefore, Specification 1.12 (which allows Frequency extensions) does not apply[10].

Maintaining containment air locks operable requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. The surveillance requirement reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program. Thus, Specification 1.12 (which allows Frequency extensions) does not apply. During normal plant operation, containment personnel lock door seals are continuously pressurized after each closure by the WCCPPS. Whenever containment integrity is required, verification is made that seals repressurize properly upon closure of an air lock door. The verification meets the intent of the 10 CFR 50 Appendix J requirements.[8]



Amendment No. 77, 174,

TSCR 98-043

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.2:
## "Containment Air Locks"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.6.2 - Containment Air Locks

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.6.A.1 specifies that containment integrity (as defined in CTS 1.10) shall not be violated; and, CTS 1.10.5 specifies that both doors in each personnel air lock must be "properly closed." Additionally, CTS 1.10.5 specifies that the air lock may be used for entry, egress or maintenance, at which time at least one air lock door shall be closed.

ITS 3.6.2 maintains the requirements in CTS 1.10.5 and includes the clarification that two (versus the less specific "each" in the CTS) air locks must be operable. Additionally, ITS 3.6.2 clarifies the ambiguous term "properly closed" by requiring that the air lock is Operable with the associated ITS Bases defining air lock Operability to require that the air lock interlock mechanism must be Operable. The air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be Operable. Finally, the statement in CTS 1.10.5 that the air lock may be used for its intended purpose (i.e., entry, egress or maintenance as long as at least one air lock door is closed) is explained in the ITS Bases. The Bases explains that the air lock safety function is met with one closed door but that both doors are kept closed when the air lock is not being used for normal entry into and exit from containment or for maintenance on the doors or the airlock.

The ITS maintains all existing requirements by dividing the containment Operability requirements into four separate LCOs: ITS 3.6.1 which requires that the containment is Operable; ITS 3.6.2 which requires that the containment air locks are Operable; ITS 3.6.3 which requires that each containment isolation valve is Operable; and, ITS LCO 3.6.9 which requires that IVSW is Operable. In conjunction with this change, the CTS definition of Containment Integrity is deleted because it contains information that is more appropriately contained in the ITS LCOs (and SRs) which establish the requirements for containment integrity and the Bases associated with these LCOs and SRs. This reorganization ensures that appropriate LCOs are recognized for any Condition and that appropriate Required Actions are implemented.

This reorganization of requirements is an administrative change with no impact on safety because the ITS requirements are reasonable interpretations of the existing requirements, except for the specific changes identified and justified in the discussion of changes for each LCO addressing containment issues.

A.4 CTS 3.6.A.1 and CTS 3.6.A.3 specify the Applicability for containment integrity as whenever the reactor is above cold shutdown. ITS 3.6.2 maintains this Applicability by requiring that Containment is Operable in Modes 1, 2, 3 and 4 (i.e., whenever the reactor is above cold shutdown). This is an administrative change with no impact on safety because there is no change to the CTS Applicability.

A.5 CTS 3.6.A.3 specifies that if the containment integrity requirements are not met (i.e., an air lock is not Operable which includes one door not Operable), then containment integrity shall be restored within one hour. ITS 3.6.2, Required Actions A.1 (one of the two doors in an air lock not Operable), B.1 (air lock interlock mechanism inoperable), and C.2 (air lock inoperable for reasons other than A or B), require verification within one hour that at least one door in the affected air lock is closed. Verification that at least one air lock door is closed ensures that containment integrity is restored (except in Condition C which is addressed in ITS 3.6.2, DOC L.2). This is an administrative change with no impact on safety because the ITS requirement (to verify at least one air lock door is closed within 1 hour) is a reasonable interpretations of the existing requirements.

A.6 The Actions for ITS 3.6.2, Containment Air Locks, are preceded by a Note (2) that specifies: "Separate Condition entry is allowed for each air lock." This allowance provides explicit recognition that the ITS is designed to allow completely separate re-entry into any Condition for each air lock addressed by the Condition including separate tracking of Completion Times based on this re-entry. This allowance is consistent with an unstated assumption in the CTS. Therefore, the addition of this Note is an administrative change with no impact on safety.

A.7 The Actions for ITS 3.6.2, Containment Air Locks, are preceded by a Note (3) that clarifies that the applicable Conditions and Required Actions of LCO 3.6.1. "Containment," are applicable when air lock leakage results in exceeding the overall containment leakage rate. This clarification is not needed in CTS 3.6.A because CTS 3.6.A requirements and associated Actions apply to the containment, the air locks and the containment isolation valves. This clarification is needed in ITS because the ITS uses separate LCOs for the containment (LCO 3.6.1), airlocks (LCO 3.6.2), and containment isolation valves (LCO 3.6.3). This is an administrative change with no impact on safety because there is no change to the existing requirements.

A.8 ITS LCO 3.6.2, Required Actions A.1 (one of the two doors in an air lock not Operable) and B.1 (air lock interlock mechanism inoperable) are modified by Note directing that the associated Required Actions are not applicable if both doors in the same air lock are inoperable and Condition C is entered. This Note is needed because Required Actions C.1 and C.2 are the appropriate remedial actions if both doors in the same air lock are inoperable and an Operable door is not available to be closed to ensure containment integrity is maintained. However, Note 1 to Required Actions A.1 and B.1 is constructed to be consistent with the ITS convention of entering all LCO Conditions that apply; therefore, the exception provided by Note 1 does not affect tracking the Completion Time from the initial entry into Condition A and/or B; only the requirement to comply with the Required Actions A and/or B when both airlock doors are inoperable. The clarification of the intent of ITS LCO 3.6.2 provided by Note 1 to Actions A.1 and B.1 is an administrative change with no impact on safety because the Notes are consistent with a reasonable interpretation of existing requirements.

A.9 CTS 4.4.D requires that air locks be tested in accordance with the Containment Leakage Rate Testing Program. ITS SR 3.6.2.1 maintains this requirement with additional guidance provided in SR 3.6.2.1, Note 2, (as modified by TSTF-52 (WOG-42), Rev 0) that results are evaluated against acceptance criteria applicable to SR 3.6.1.1, Containment Leakage Rate Testing Program. SR 3.6.2.1 ensures that acceptance criteria for air lock testing, listed in the ITS 5.5.15, Containment Leakage Rate Testing Program, is met. SR 3.6.2.1, Note 2, is added to ensure that air lock leakage is also included in determining the overall containment leakage rate which is determined by ITS SR 3.6.1.1. This is an administrative change with no impact on safety because it is a clarification that ensures proper interpretation of the existing requirements.

A.10 CTS 4.4.D requires that air locks be tested in accordance with the Containment Leakage Rate Testing Program. ITS SR 3.6.2.1 maintains this requirement with additional guidance in ITS SR 3.6.2.1, Note 1. This Note specifies that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This change is acceptable because either air lock door is capable of providing a fission product barrier in the event of a DBA. This is an administrative change with no impact on safety because SR 3.6.2.1, Note 1, is consistent with a reasonable interpretation of the existing requirement.

MORE RESTRICTIVE

M.1 CTS 3.6.A.3 specifies that if the containment integrity requirements are not met (i.e., an air lock is not Operable), then containment integrity shall be restored within one hour. Under the same conditions, ITS 3.6.2, Required Actions A.1 and B.1, maintain this requirement (see ITS 3.6.2, DOC A.5); however, ITS 3.6.2, Required Actions A.2, A.3, B.2, B.3 and associated Notes, is more restrictive by requiring that the Operable door in the affected air lock must be locked shut within 24 hours and verified locked closed every 31 days thereafter unless the air lock door is in a high radiation area, in which case, administrative verification is acceptable.

This change is needed to provide an appropriate level of assurance that containment integrity is maintained when one air lock door and/or the interlock mechanism are inoperable. The allowance permitting air lock doors in high radiation areas to be verified locked closed by administrative means is acceptable because access to these areas is restricted which significantly reduces the probability of misalignment of the door after it has been verified to be locked in the proper position. This more restrictive change is acceptable because having the Operable airlock door locked shut when the other door and/or the interlock mechanism are inoperable provides a very high degree of assurance that containment integrity is maintained with no impact on plant operation or personal safety related to the reduced accessibility to the containment.

M.2 CTS 3.6.A and CTS 1.10 do not establish any explicit requirements for the Operability of the containment air lock interlock mechanism. Consequently, CTS 1.10.5 is interpreted as allowing entry, egress or maintenance without the interlock as long as at least one air lock door remains closed. Under the same conditions, ITS 3.6.2, Condition B, requires compensatory actions for an airlock with an inoperable interlock mechanism equivalent to the compensatory actions for an inoperable airlock door. In conjunction with this change, ITS 3.6.2, Required Action B.1, Note 2, allows entry into and exit from containment via an airlock with an inoperable interlock only if performed under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

This more restrictive change is needed to provide assurance that containment integrity is maintained during air lock operation when the door interlock is not Operable. This more restrictive change is acceptable because use of a dedicated individual to monitor air lock operation when the interlock is inoperable provides a very high degree of assurance that containment integrity is maintained with no impact on plant operation or personal safety.

M.3 CTS 3.6.A and CTS 1.10 do not establish any explicit requirements for the Operability or testing of the door interlock mechanism on containment airlocks. ITS SR 3.6.2.2 (as modified by TSTF-17 (WOG-33), Revision 1) is added to require verification of the Operability of each air lock interlock mechanism every 24 months. This additional requirement is needed because the air lock door interlock feature supports containment Operability while the air lock is being used for containment ingress and egress. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not occur. The 24 month Frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

LESS RESTRICTIVE

L.1 CTS 3.6.A and CTS 1.10 do not establish any explicit allowance for containment ingress or egress through an air lock with an inoperable door; therefore, ingress or egress through an air lock with an inoperable inner door results in a breach of containment and CTS 3.6.A.3 is applicable. Under the same conditions, ITS LCO 3.6.2, Actions Note 1, specifies that entry and exit is permissible to perform repairs on the affected air lock components (although the Bases specify that entry via the Operable air lock is preferred). Additionally, if both air locks have inoperable doors, ITS LCO 3.6.2, Required Action A.1, Note 2, specifies that entry and exit for any reason is permissible for a period of 7 days beginning when the second air lock becomes inoperable.

These Notes are needed because they provide explicit recognition that LCO Required Actions are not appropriate for a breach of containment integrity that is planned, closely controlled and of very short duration such as occurs when an air lock door is inoperable. Additionally, the same activity is permissible under CTS 3.6.A.3 with a 1 hour limit to restore containment integrity. These allowances are acceptable because of the following: 1) the same activity is permissible under CTS 3.6.A.3 with a 1 hour limit to restore containment integrity; and, 2) there is a low probability of an event that could pressurize the containment during the short time that the Operable door is expected to be open. Therefore, this change has no impact on safety.

L.2 CTS 3.6.A.3 specifies that if the containment integrity requirements are not met (i.e., air lock not Operable), then containment integrity shall be restored within one hour. ITS 3.6.2 differentiates between Conditions that clearly do not result in a breach of containment integrity (inoperability of a single air lock door (ITS 3.6.2, Condition A) and/or an inoperable interlock (ITS 3.6.2, Condition B)) and Conditions that have the potential for exceeding the overall containment leakage rate limit established in LCO 3.6.1. (See 3.6.2, DOCs A.5 and M.1 for Conditions that do not result in a breach of containment integrity.) Instead of requiring a shutdown within one hour for conditions that have the potential for exceeding the overall containment leakage rate limit per LCO 3.6.1, ITS 3.6.2, Condition C and associated Required Actions, requires the following: immediate evaluation of previous combined leakage rates in conjunction with current air lock test results; verification within one hour that a door in the affected air lock is closed; and, initiation of a reactor shutdown within 24 hours if at least one air lock door is not restored to Operable (i.e., containment integrity intact with at least one Operable door).

The addition of Required Actions C.1, C.2 and C.3 establishes a new Completion Time of 24 hours to evaluate if an inoperable airlock causes overall containment leakage rate limit established in LCO 3.6.1 is exceeded before reactor shutdown is required. This change is acceptable because LCO 3.6.2, Required Action C.1 and Actions Note 3, implicitly requires initiation of a reactor shutdown in accordance with LCO 3.6.1 within one hour of any reasonable determination that overall containment leakage rate limits per LCO 3.6.1 are exceeded. Otherwise, an evaluation lasting no more than 24 hours is acceptable because it is overly conservative to assume that overall containment leakage rate limit is not met even if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. LCO 3.6.2, Required Action C.3, establishes the 24 hour limit for resolution of any uncertainty related to the affect of air lock Operability on overall containment leakage. Finally, Required Action C.2, the requirement that one door in the affected containment air lock be verified closed within 1 hour, is consistent with CTS 3.6.A.3 requirements and LCO 3.6.1, which require that containment be restored to Operable status within 1 hour.

This change does not have a significant impact on safety because a prompt reactor shutdown is still required if it is apparent that the overall containment leakage rate limit is not met. However, in situations where the overall containment leakage rate may still be within limits, an evaluation limited to 24 hours is justified because the probable outcome is that leakage is within limits or exceeded only marginally and the low probability of an event during the 24 hour evaluation period.

REMOVED DETAIL

None

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.2:
## "Containment Air Locks"

## PART 4:

## No Significant Hazards Considerations
## for
## Changes between CTS and ITS
## that are
## Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.6.2 - Containment Air Locks

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change creates an allowance permitting ingress and egress from the containment using an air lock with an inoperable door to perform repairs on the affected air lock components (although the Bases specify that entry via the Operable air lock is preferred). Additionally, this change creates an allowance permitting ingress and egress from the containment to perform any required task using an air lock with an inoperable door for a period of 7 days beginning with the inoperability of a door in the second air lock. Without this allowance, the Conditions and associated Required Actions for a breach of containment would be applicable during each containment entry or exit. This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because of the following: the same activity is permissible under CTS 3.6.A.3 with a 1 hour limit to restore containment integrity; and, there is a low probability of an event that could pressurize the containment during the short time that the Operable door is expected to be open.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in margin of safety because of the following: the same activity is permissible under CTS 3.6.A.3 with a 1 hour limit to restore containment integrity; and, there is a low probability of an event that could pressurize the containment during the short time that the Operable door is expected to be open.

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LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change differentiates between Conditions that clearly do not result in a breach of containment integrity (inoperability of a single air lock door (ITS 3.6.2, Condition A) and/or an inoperable interlock (ITS 3.6.2, Condition B)) and Conditions that have the potential for exceeding the overall containment leakage rate limit per LCO 3.6.1. Instead of requiring a shutdown within one hour for conditions that have the potential for exceeding the overall containment leakage rate limit per LCO 3.6.1, ITS 3.6.2, Condition C and associated Required Actions, require the following: immediate evaluation of previous combined leakage rates in conjunction with current air lock test results; verification within one hour that a door in the affected air lock is closed; and, initiation of a reactor shutdown within 24 hours if the air lock is not restored to Operable.

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because LCO 3.6.2, Required Action C.1 and Actions Note 3, implicitly require initiation of a reactor shutdown in accordance with LCO 3.6.1 within one hour of any reasonable determination that overall containment leakage rate limits are exceeded. Otherwise, an evaluation lasting no more than 24 hours is acceptable because it is overly conservative to assume that overall containment leakage rate limit is not met even if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. LCO 3.6.2, Required Action C.3, establishes the 24 hour limit for resolution of any uncertainty related to the affect of air lock Operability on overall containment leakage. Finally, Required Action C.2, the requirement that one door in the affected containment air lock be verified closed within 1 hour, is consistent with CTS 3.6.A.3 requirements and LCO 3.6.1, which require that containment be restored to Operable status within 1 hour. Therefore, a prompt reactor shutdown is still required if it is apparent that the overall containment leakage rate limit is not met. However, in situations where the overall containment leakage rate may still be within limits, an evaluation limited to 24 hours is justified because the probable outcome is that leakage is within limits or exceeded only marginally.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because initiation of a reactor shutdown within one hour of any reasonable determination that overall containment leakage rate limits are exceeded. Additionally, the requirement that one door in the affected containment air lock be verified closed within 1 hour, is consistent with CTS 3.6.A.3 requirements and LCO 3.6.1, which require that containment be restored to Operable status within 1 hour.

# Indian Point 3
# Improved Technical Specifications (ITS)
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# Technical Specification 3.6.2:
# "Containment Air Locks"

## PART 5:

## NUREG-1431
## Annotated to show differences between NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.6.2

This ITS Specification is based on NUREG-1431 Specification No. **3.6.2**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| WOG-033 | **017 R0** | EXTENSION OF TESTING FREQUENCY OF CONTAINMENT AIRLOCK INTERLOCK MECHANISM FROM 184 DAYS TO 24 MONTHS | See Next Rev. | See Next Rev. | **N/A** |
| WOG-033 R1 | **017 R1** | EXTENSION OF TESTING FREQUENCY OF CONTAINMENT AIRLOCK INTERLOCK MECHANISM FROM 184 DAYS TO 24 MONTHS | Approved by NRC | Incorporated | **T.1** |
| WOG-042 | **052** | IMPLEMENT 10 CFR 50, APPENDIX J, OPTION B | TSTF to Rewrite | TSTF is CLB | **N/A** |

<CTS>

## 3.6 CONTAINMENT SYSTEMS

### 3.6.2 Containment Air Locks ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~

<1.10.5> <3.6.A.1> <DOC A.3>

LCO 3.6.2 [Two] containment air lock[s] shall be OPERABLE.

<3.6.A.1> <DOC A.4>

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

<DOC L.1> 1. Entry and exit is permissible to perform repairs on the affected air lock components.

<DOC A.6> 2. Separate Condition entry is allowed for each air lock.

<DOC A.7> <DOC L.2> 3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate.

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<3.6.A.3> <DOC A.8> <DOC L.1>

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more containment air locks with one containment air lock door inoperable. | -----NOTES-----<br>1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered.<br>2. Entry and exit is permissible for 7 days under administrative controls [if both air locks are inoperable].<br>------------------- | |
| | | (continued) |



Typical

ACTIONS

| | CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|---|
| <3.6.A.3> <DOC A.5> | A. (continued) | A.1 | Verify the OPERABLE door is closed in the affected air lock. | 1 hour |
| | | <u>AND</u> | | |
| <DOC M.1> | | A.2 | Lock the OPERABLE door closed in the affected air lock. | 24 hours |
| | | <u>AND</u> | | |
| <DOC M.1> | | A.3 | --------NOTE--------- Air lock doors in high radiation areas may be verified locked closed by administrative means. --------------------- | |
| | | | Verify the OPERABLE door is locked closed in the affected air lock. | Once per 31 days |

(continued)

ACTIONS (continued)

| <CTS> | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <3.6.A.3><br><DOC M.2><br><DOC A.8><br><br><DOC M.2> | B. One or more containment air locks with containment air lock interlock mechanism inoperable. | ------------NOTES------------<br>1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered.<br><br>2. Entry and exit of containment is permissible under the control of a dedicated individual.<br>------------------------------ | |
| <3.6.A.3><br><DOC A.5> | | B.1 Verify an OPERABLE door is closed in the affected air lock. | 1 hour |
| | | <u>AND</u> | |
| <DOC M.1> | | B.2 Lock an OPERABLE door closed in the affected air lock. | 24 hours |
| | | <u>AND</u> | |
| <DOC M.1> | | B.3 --------NOTE---------<br>Air lock doors in high radiation areas may be verified locked closed by administrative means.<br>---------------------<br><br>Verify an OPERABLE door is locked closed in the affected air lock. | Once per 31 days |

(continued)

ACTIONS (continued)

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <3.6.A.3> <DOC L.2> | C. One or more containment air locks inoperable for reasons other than Condition A or B. | C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1. | Immediately |
| | | <u>AND</u> | |
| <3.6.A.3> <DOC A.5> | | C.2 Verify a door is closed in the affected air lock. | 1 hour |
| | | <u>AND</u> | |
| <DOC L.2> | | C.3 Restore air lock to OPERABLE status. | 24 hours |
| <3.6.A.3> | D. Required Action and associated Completion Time not met. | D.1 Be in MODE 3. | 6 hours |
| | | <u>AND</u> | |
| | | D.2 Be in MODE 5. | 36 hours |

SURVEILLANCE REQUIREMENTS

| | | SURVEILLANCE | FREQUENCY | |
|---|---|---|---|---|
| <CTS> | | | | |
| <4.4.D><br><DOC A.10><br><br><DOC A.9><br>applicable to | SR 3.6.2.1 | -----NOTES-----<br>1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.<br><br>2. Results shall be evaluated against acceptance criteria of SR 3.6.1.1 ~~in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.~~<br>----- | the integrated leak rate test. | |
| <4.4.D><br>the Containment Leakage Rate Testing Program | | Perform required air lock leakage rate testing in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~<br><br>~~The acceptance criteria for air lock testing are:~~<br><br>~~a. Overall air lock leakage rate is ≤ [0.05 $L_a$] when tested at ≥ $P_a$.~~<br><br>~~b. For each door, leakage rate is ≤ [.01 $L_a$] when tested at ≥ [ psig].~~ | ~~-----NOTE----- SR 3.0.2 is not applicable -----~~<br><br>In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ | CLB.1 |
| <DOC H.3> | [SR 3.6.2.2 | ~~-----NOTE-----<br>Only required to be performed upon entry or exit through the containment air lock.<br>-----~~<br><br>Verify only one door in the air lock can be opened at a time. | 24 months<br>~~184 days~~] | T.1 |

## B 3.6 CONTAINMENT SYSTEMS

## B 3.6.2 Containment Air Locks ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~

## BASES

**BACKGROUND**

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

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~~Each air lock is nominally a right circular cylinder, 10 ft in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).~~

Each personnel air lock is provided with limit switches on both doors that provide control room indication ~~of door position~~. ~~Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated.~~

when an airlock door is not fully closed.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

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NUREG-1431 Markup Inserts
ITS SECTION 3.6.2 - Containment Air Locks

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Each air lock is a cylinder with a door at each end. One of the two air locks is designed as a part of the containment structure and the other is designed as an integral part of the containment equipment hatch but otherwise the two air locks function identically. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY.

Each air lock door and the equipment hatch is designed with double gasketed seals to permit pressurization between the gaskets. The double gasketed seals are normally continuously pressurized above accident pressure. Finally, to effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door) and local leakage rate testing capability is available to ensure containment integrity is being maintained.

The doors are interlocked to prevent simultaneous opening of the inner and outer door. This interlock is a requirement for OPERABILITY. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident ~~(Ref. 2)~~. In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.1]% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as $L_a$ = [0.1]% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure ~~($P_a$ = [14.4] psig)~~ following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

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The containment air locks satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into ~~and~~ or exit from containment.

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APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the

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The peak pressure following a DBA is specified in Specification 5.5.15, Containment Leakage Rate Testing Program.

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Pressurization of air lock seals is not required for air lock OPERABILITY. However, 10 CFR 50, Appendix J, Section III.2.b(iii), specifies that air locks opened during periods when containment integrity is required must be tested within 3 days after being opened. However, for air lock doors having testable seals, testing the seals (i.e., verification that seals re-pressurize to the required pressure after an air lock door is closed) fulfills the 3-day test requirements. Therefore, the status of air lock seals has the potential to affect air lock OPERABILITY.

BASES

APPLICABILITY (continued)

probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

When the inner door is inoperable

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

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BASES

ACTIONS (continued)

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

(continued)

BASES

ACTIONS (continued)

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

unless Condition C is exited in accordance with LCO 3.0.2 (i.e., one door is made Operable

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

PA.1

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), ~~as modified by approved exemptions~~. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is ~~required by Appendix J (Ref. 1), as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~

in accordance with Specification 5.5.15, Containment Leakage Rate Testing Program.

CLB.1

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the overall containment leakage rate.

that is applicable to

Combined Type B and C

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given ~~that the interlock mechanism is only~~ challenged when the ~~containment air lock door is opened~~, this test is only required to be performed ~~upon entering or exiting a containment air lock but is not required more frequently~~

not normally

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used for entry and exit ~~(procedures require strict adherence to single door opening)~~.

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every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

~~than every 184 days~~. The ~~184 day~~ Frequency is based on engineering judgment and is considered adequate ~~in view of other indications of door and interlock mechanism status available to operations personnel~~.

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REFERENCES

1. 10 CFR 50, Appendix J.

2. FSAR, Section [~~6.2~~]. 6.6

NUREG-1431 Markup Inserts
ITS SECTION 3.6.2 - Containment Air Locks

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given that the interlock is not normally challenged during the use of the airlock

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.2:
# "Containment Air Locks"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

## JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
## ITS SECTION 3.6.2 - Containment Air Locks

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 This change maintains IP3 current licensing basis related to the use of 10 CFR 50, Appendix J, Option B, for containment leak rate testing which was approved on June 17, 1997 as part of Amendment 174. This change is based on Generic Change TSTF-52 (WOG-42), Revision 0, which is currently being reviewed by the NRC.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DB.2 Descriptions of the air locks and interlock mechanisms are revised to be consistent with the IP3 design as described in the IP3 FSAR.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-17 (WOG-33), Rev 1, which extends the testing frequency of containment airlock interlock mechanisms from 184 days to 24 months. This change is needed because testing of the airlock interlock mechanism is accomplished through having one door not completely engaged in the closed position, while attempting to open the second door. Failure of this surveillance effectively results in a loss of containment integrity. Procedures and training do not allow this interlock to be challenged for ingress and egress. This change is acceptable because when an air lock is opened during times the interlock is required, the operator first verifies that one door is completely shut and the door seals pressurized before attempting to open the other door. Therefore, the interlock is not challenged except during actual testing of the interlock. Consequently, it should be sufficient to ensure proper operation of the interlock by testing the interlock on a 24 month interval. Historically, this interlock verification has had its frequency chosen to coincide with the frequency of the overall airlock leakage test. According to 10 CFR 50, Appendix J, Option A, this frequency is once per 6 months. However, Appendix J, Option B, allows for an extension of the overall airlock leakage test frequency to a maximum of 30 months. Finally, a 24 month Frequency allows the interlock to be tested in a Mode where the interlock is not required.

DIFFERENCES FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.3:
# "Containment Isolation Valves"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.6 CONTAINMENT SYSTEMS

### 3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

------------------------------------NOTES------------------------------------

1. Penetration flow path(s) except for 36 inch purge valve flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

5. Enter applicable Conditions and Required Actions of LCO 3.6.9, "Isolation Valve Seal Water (IVSW) System," when required IVSW supply to a penetration flowpath is inoperable.

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ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. ---------NOTE--------- Only applicable to penetration flow paths with two or more containment isolation valves. --------------------- <br><br> One or more penetration flow paths with one containment isolation valve inoperable. | A.1 | Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. | 4 hours |
| | <u>AND</u> | | |
| | A.2 | --------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. -------------------- <br><br> Verify the affected penetration flow path is isolated. | Once per 31 days for isolation devices outside containment <br><br> <u>AND</u> <br><br> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| B. ---------NOTE---------<br>Only applicable to penetration flow paths with two or more containment isolation valves.<br>---------------------<br><br>One or more penetration flow paths with two containment isolation valves inoperable. | B.1 | Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. | 1 hour |
| C. ---------NOTE---------<br>Only applicable to penetration flow paths with only one containment isolation valve and a closed system.<br>---------------------<br><br>One or more penetration flow paths with one containment isolation valve inoperable. | C.1 | Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. | 72 hours |
| | <u>AND</u> | | |
| | C.2 | --------NOTE---------<br>Isolation devices in high radiation areas may be verified by use of administrative means.<br>--------------------<br><br>Verify the affected penetration flow path is isolated. | Once per 31 days |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| D. Required Action and associated Completion Time not met. | D.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | D.2 | Be in MODE 5. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.3.1 | Verify each 36 inch purge supply and exhaust isolation valve is sealed closed. | 31 days |
| SR 3.6.3.2 | Verify each 10 inch pressure relief isolation valve is closed, except when these valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. | 31 days |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.3.3 | -------------------NOTE-------------------<br>Valves and blind flanges in high radiation areas may be verified by use of administrative controls.<br>------------------------------------------<br><br>Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls. | 31 days |
| SR 3.6.3.4 | -------------------NOTE-------------------<br>Valves and blind flanges in high radiation areas may be verified by use of administrative means.<br>------------------------------------------<br><br>Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls. | Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.3.5 | Verify the isolation time of each power operated and each automatic power operated containment isolation valve is within limits. | In accordance with the Inservice Testing Program |
| SR 3.6.3.6 | Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal. | 24 months |
| SR 3.6.3.7 | Verify each 10 inch containment pressure relief line isolation valve is blocked to restrict valve opening to ≤ 60 degrees. | 24 months |
| SR 3.6.3.8 | Verify the combined leakage rate for all containment bypass leakage paths in accordance with the Containment Leakage Rate Testing Program. | In accordance with the Containment Leakage Rate Testing Program |

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.3 Containment Isolation Valves

BASES

### BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the containment purge supply and exhaust isolation valves (FCV-1170, FCV-1171, FCV-1172, and FCV-1173) and the containment pressure relief isolation valves (PCV-1190, PCV-1191, and PCV-1192) close when high radiation levels are detected by the Containment Air Particulate Monitor (R-11) or Containment Radioactive Gas Monitor (R-12). Containment purge and containment pressure relief are also isolated when high radiation levels are detected in the plant vent. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment

BASES

BACKGROUND (Continued)

in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

Containment Purge System (36 inch purge valves)

The Containment Purge System, consisting of purge supply and exhaust isolation valves FCV-1170, FCV-1171, FCV-1172, and FCV-1173, operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size, the 36 inch purge valves are not qualified for automatic closure from their open position under DBA conditions. Therefore, the 36 inch purge valves must be maintained sealed closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Containment Pressure Relief Line (10 inch valves)

The Containment Pressure Relief Line consisting of pressure relief isolation valves PCV-1190, PCV-1191, and PCV-1192, operates to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access, and

b. Equalize internal and external pressures.

Since the valves used in the Containment Pressure Relief Line System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4. Containment pressure relief line isolation valve opening is limited by mechanical stops so that

BASES

BACKGROUND (Continued)

opening angle is limited to an angle at which analysis indicates the valve will operate against containment accident pressures. Additionally, pressure relief isolation valve opening must be limited to the time necessary for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.

The containment pressure relief line is isolated during CORE ALTERATIONS and movement of irradiated fuel inside containment in accordance with requirements established in LCO 3.9.3, Containment Penetrations.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBA that results in a release of radioactive material within containment is a loss of coolant accident (LOCA) (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves are minimized. The safety analyses assume that the 36 inch purge valves are sealed closed at event initiation.

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, $L_a$. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The containment purge supply and exhaust isolation valves (FCV-1170, FCV-1171, FCV-1172, and FCV-1173) may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment purge valves due to failure in the control circuit associated with each valve. Again, the purge system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The term sealed closed, as applied to containment isolation valves, is not intended to describe leak tightness. Sealed closed isolation valves must be under administrative controls that assure the valve cannot be inadvertently opened. Administrative controls includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator (Ref. 3). Sealed closed barriers include blind flanges and sealed closed isolation valves including closed manual valves, closed remote-manual valves, and closed automatic valves which remain closed after a loss-of-coolant accident. Sealed closed barriers may be used in place of any automatic isolation valve.

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36.

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 36 inch purge valves must be maintained sealed closed.

BASES

LCO (continued)

The valves covered by this LCO are listed in the FSAR (Ref. 2). The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact (Ref. 3).

Manually operated containment isolation valves on essential lines that are required to be open, at least for a time, during post accident conditions are OPERABLE if they can be closed in accordance with design assumptions. Essential lines are those lines required to mitigate an accident, or which, if unavailable, could increase the magnitude of the event. Also, those lines which, if available, would be used in the short term (24 to 36 hours) to restore the plant to normal operation following an event which has resulted in containment isolation (Ref. 4).

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, Containment Penetrations.

ACTIONS

The ACTIONS are modified by Note 1, which allows penetration flow paths that are isolated in accordance with Required Actions, except for 36 inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous

ACTIONS (continued)

communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls.

Note 2 has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by Note 3, which ensures appropriate remedial actions are taken if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event containment isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

The ACTIONS are further modified by Note 5, which ensures appropriate remedial actions are taken if required IVSW supply to a penetration flowpath is inoperable. Specifically, Note 5 directs entry into the applicable Conditions and Required Actions of LCO 3.6.9.

<u>A.1 and A.2</u>

In the event one containment isolation valve in one or more penetration flow paths is inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be

BASES

ACTIONS

A.1 and A.2 (continued)

adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured (Ref. 3). For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. This action involves verification, through a system walkdown, that isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment e.g., one of the three containment pressure relief isolation valves, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with

BASES

ACTIONS

A.1 and A.2 (continued)

two or more containment isolation valves. Although most penetrations have two containment isolation valves, the term "two or more" is used so that Condition A includes the pressure relief line penetration which has three valves in series. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

B.1

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment

BASES

ACTIONS

B.1 (continued)

isolation valves. Although most penetrations have two containment isolation valves, the term "two or more" is used so that Condition B includes the pressure relief line penetration which has three valves in series. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier, other than the closed system that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange (Ref. 3). A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

BASES

ACTIONS

C.1 and C.2 (continued)

Required Action C.2 is modified by a Note that applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1 and D.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 36 inch containment purge supply and exhaust isolation valve (FCV-1170, FCV-1171, FCV-1172, and FCV-1173) is required to be verified sealed closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. The term sealed closed, as applied to containment isolation valves, is not intended to describe leak tightness. Sealed closed isolation valves must be under administrative controls that assure the valve cannot be inadvertently opened. Administrative controls includes mechanical devices to seal or

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1 (continued)

lock the valve closed, or to prevent power from being supplied to the valve operator (Ref. 3). Sealed closed barriers include blind flanges and sealed closed isolation valves including closed manual valves, closed remote-manual valves, and closed automatic valves which remain closed after a loss-of-coolant accident.

The Frequency is a result of an NRC initiative, Generic Issue B-24 (Ref. 5), related to containment purge valve use during plant operations.

SR 3.6.3.2

This SR ensures that the containment pressure relief line isolation valves (PCV-1190, PCV-1191, and PCV-1192) are closed as required or, if open, open for an allowable reason. If a containment pressure relief line isolation valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the containment pressure relief line isolation valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The containment pressure relief line isolation valves are capable of closing in the environment following a LOCA as long as valve opening angle is limited in accordance with SR 3.6.3.7. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3 (continued)

of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed or otherwise secured in the closed position because these valves were verified to be in the correct position when locked, sealed or otherwise secured.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.4 (continued)

that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed or otherwise secured in the closed position because these valves were verified to be in the correct position when locked sealed or otherwise secured.

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses as specified in the FSAR. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.6

Automatic power operated containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.6 (continued)

transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.7

Verifying that each containment pressure relief line isolation valve, PCV-1190, PCV-1191, and PCV-1192, is blocked to restrict valve opening to ≤ 60 degrees. This verification is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the pressure releif line valves must close to maintain containment leakage within the values assumed in the accident analysis. The 24 month Frequency is appropriate because the blocking devices can be removed only when plant is in MODE 5 or 6.

SR 3.6.3.8

This SR ensures that the combined leakage rate of all containment leakage paths is less than or equal to the specified leakage rate for those paths that are not sealed by the Isolation Valve Seal Water System or sealed by the RHR system or sealed by the service water system. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves.

BASES

SURVEILLANCE REQUIRMENTS

SR 3.6.3.8 (continued)

This testing is performed in accordance with the requirements, Frequency and acceptance criteria established in Specification 5.5.15, Containment Leakage Rate Testing Program. This program was established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by IP3 specific approved exemptions. This program conforms to guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995." In the event containment isolation valve leakage results in exceeding the overall containment leakage rate, entry into the applicable Conditions and Required Actions of LCO 3.6.1 is required.

REFERENCES

1. FSAR, Section 14.
2. FSAR, Section 6.
3. Standard Review Plan Section 6.2.4, Item II.3.f.
4. FSAR, Section 5.2.
5. Generic Issue B-24.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.6.3:
# "Containment Isolation Valves"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 1-4 | 34 TSCR 97-070 | 34 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| 3.6-1 | 86 TSCR 97-070 | 86 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| 3.6-2 | 3-16-95 TSCR 97-175 | 3-16-95 TSCR 97-175 | IPN 97-175 | Changes to Bases Pages | |
| 3.6-3 | 8-30-95 TSCR 97-070 | 8-30-95 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| T 3.6-1 | 152 TSCR 97-070 | 152 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| T 4.1-3(1) | 178 TSCR 97-156, 98-043 | 178 TSCR 97-156, 98-043 | IPN 98-043 | Instrument Channel Surveillance Intervals Extended to 24 Months | Incorporated |
| T 4.1-3(1) | 178 TSCR 97-156, 98-043 | 178 TSCR 97-156, 98-043 | IPN 97-156 | SR Freq for Main Turbine Stop and Control Valves | Incorporated |
| 4.4-4 | 174 | 174 | No TSCRs | No TSCRs for this Page | N/A |
| 4.4-5 | 174 | 174 | No TSCRs | No TSCRs for this Page | N/A |

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.3:
## "Containment Isolation Valves"

| | | | | | |
|---|---|---|---|---|---|
| **4.4-7** | **174 TSCR 98-043** | **174 TSCR 98-043** | **IPN 98-043** | Instrument Channel Surveillance Intervals Extended to 24 Months | **Incorporated** |
| **4.4-8** | **174 TSCR 98-043** | **174 TSCR 98-043** | **IPN 98-043** | Instrument Channel Surveillance Intervals Extended to 24 Months | **Incorporated** |
| **4.4-9** | **174** | **174** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **4.4-10** | **174 TSCR 98-043** | **174 TSCR 98-043** | **IPN 98-043** | Instrument Channel Surveillance Intervals Extended to 24 Months | **Incorporated** |
| **T 4.4-1(1)** | **102** | **102** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **T 4.4-1(2)** | **152** | **152** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **T 4.4-1(3)** | **102** | **102** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **T 4.4-1(4)** | **102** | **102** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **T 4.4-1(5)** | **105 TSCR 97-070** | **105 TSCR 97-070** | **IPN 97-070** | Clarification of Containment Integrity | **Incorporated** |
| **T 4.4-1(6)** | **174** | **174** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **T 4.4-1(7)** | **102** | **102** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **4.13-1** | **175** | **175** | **No TSCRs** | No TSCRs for this Page | **N/A** |
| **4.13-2** | **131** | **131** | **No TSCRs** | No TSCRs for this Page | **N/A** |

~~1.9.2~~ Instrument Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating actions.

~~1.9.3~~ Instrument Channel Calibration

SEE ITS 1.0

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the channel functional test.

~~1.9.4~~ Logic Channel Functional Test

The operation of relays or switch contacts, in all the combinations required, to produce the required output.

~~1.10~~ CONTAINMENT INTEGRITY

LCO 3.6.3 Containment integrity is defined to exist when:

~~1.10.1~~ LCO 3.6.3 Actions Note 1 Non-automatic containment isolation valves (Table 3.6-1) are closed or may be opened under administrative control ~~and only as long as necessary to perform their intended function.~~ L.1 A.3

~~1.10.2~~ SR 3.6.3.3 ~~Blind flanges, that provide an isolation function which are shown in FSAR drawings, are maintained installed.~~

~~1.10.3~~ SR 3.6.3.4 ~~Any test connection, vent or drain valve that is located within the isolation boundary and is required to perform an isolation function is closed and capped (threaded) or blind flanged as shown in FSAR drawings.~~ M.3

~~1.10.4~~ SEE ITS 3.6.1 The equipment door is properly closed.

~~1.10.5~~ SEE ITS 3.6.2 Both doors in each personnel air lock are properly closed unless being used for entry, egress or maintenance, at which time at least one air lock door shall be closed.

LCO 3.6.3 1.~~10.6~~ Req. Act A.1 B.1 C.1 All automatic containment isolation valves are ~~either~~ operable ~~or in the closed position, or isolated by a closed manual~~ valve or flange that meets the same design criteria as the isolation valve.

Add Note to Conditions A, B and C A.8

L.2

A.9

A.10



Amendment No. 34,

TSCR 97-070

Add Req Act A.1 and C.1 L.2

3.6 CONTAINMENT SYSTEM

~~Applicability~~

~~Applies to the integrity of reactor containment.~~

~~Objective~~

~~To define the operating status of the reactor containment for plant operation.~~

~~Specification~~

A.2

Mode 1, 2, 3 and 4 — A.4

A. Containment Integrity

Isolation valves shall be Operable — A.3

LCO 3.6.3

1. Containment ~~integrity (as defined in 1.10) shall not be violated unless the reactor is in the cold shutdown condition~~. Those valves to be opened continuously or intermittently are under administrative control ~~and are open only as long as necessary to perform their intended function~~.

SEE ITS 3.9.1

2. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is sufficient to maintain the shutdown margin equal to or greater than the requirements of specification 3.8.D.

LCO 3.6.3

Req Act B.1

Req Act D.1-2

3. If the containment integrity requirements are not met ~~when the reactor is above cold shutdown~~, containment integrity shall be restored within one hour or the reactor shall be in the ~~hot shutdown~~ Mode 3 condition within six hours and in ~~cold shutdown condition~~ Mode 5 within the next 30 hours. — A.4 — L.2 — A.10

SEE ITS 3.6.4

B. Internal Pressure

If the internal pressure exceeds 2.5 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor shutdown.

Add Note 4 to Actions — A.5

Add Note 3 to Actions — A.7

Add Note 2 to Actions — A.6

Add Req Act A.2 and C.2 — M.1



Amendment No. 14, 66

TSCR 97-070

C. Containment Temperature

SEE ITS 3.6.5

1. The reactor shall not be taken above the cold shutdown condition unless the containment ambient temperature is greater than 50°F.

2. Containment ambient temperature shall not exceed 130°F when the reactor is above the cold shutdown condition. If the temperature is greater than 130°F, reduce the temperature to within the limit within 8 hours, or be in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

3. Containment ambient temperature as specified in 3.6.C.1 and 3.6.C.2 shall be the arithmetic average of temperatures measured at no fewer than 4 locations, at least once per 24 hours.

A.11

D. Containment ~~Vent~~ and Purge System Pressure Relief

in Mode 1, 2, 3 and 4 A.4

SR 3.6.3.7 The reactor shall not be ~~taken above the cold shutdown condition~~ unless the containment vent isolation valves (PCV - 1190, - 1191, - 1192) are closed or limited to a maximum valve opening angle of 60° (90° - full open) by mechanical means.

in Mode 1, 2, 3 and 4 A.4

SR 3.6.3.1 The reactor shall not be ~~taken above the cold shutdown condition~~ unless the containment purge supply and exhaust isolation valves (FCV - 1170, - 1171, - 1172, - 1173) are closed.

Req. Act D.1, D.2 If the above conditions cannot be met within one hour, the reactor shall be in the ~~hot shutdown condition~~ Mode 3 within six hours and in the ~~cold shutdown condition~~ Mode 5 within the next 30 hours. L.2

~~Basis~~

~~The Reactor Coolant System must be in the cold shutdown condition in order to relax containment integrity. When the Reactor Coolant System is in the cold shutdown condition, the pressurizer may have an internal temperature above 200°F for purposes of drawing and maintaining a steam bubble, provided that the reactor has been subcritical for at least 24 hours. Operation in this manner ensures that, in case of an accidental RCS coolant release under cold shutdown conditions, the ensuing offsite radiation doses will be within the limits of 10 CFR 100.~~

~~The shutdown margins are selected on the type of activities that are being carried out. The shutdown margin requirement of specification 3.8.D when the vessel head bolts are less than fully tensioned precludes criticality during refueling. When the reactor head is not to be removed, the specified cold shutdown margin of 1% Δ k/k precludes criticality in any occurrence.~~

A.1

A.1

Regarding internal pressure limitations, the containment design pressure of 47 psig would not be exceeded for a major loss-of-coolant accident or for a main steam line break accident.[1] The loss-of-coolant accident event bounds the main steam line break accident from the containment peak pressures standpoint. The initial pressure condition used in the containment analysis was 2.5 psig.[1] The containment can withstand an internal vacuum of 3 psig.[2] The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

The requirement of a 50°F minimum containment ambient temperature is to assure that the minimum service metal temperature of the containment liner is well above the NDT + 30°F criterion for the liner material. [3]

Limiting maximum containment ambient temperature will ensure that the peak accident containment pressure does not exceed the design limit of 47 psig during steamline break or loss of coolant accidents. Environmentally and seismically qualified RTDs mounted on the crane wall above the containment fan cooler units inlet are normally used for measuring containment ambient temperature. Portable temperature sensing equipment may also be used, provided the criteria of 3.6.C.3 are met.

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function. [4] During periods of normal plant operations requiring containment integrity, some of the valves on this Table will be open either continuously or intermittently depending on requirements of the particular protection, safeguards or essential service systems. Those valves to be opened are under administrative control and are open only as long as necessary to perform their intended function. Some of the valves listed in Table 3.6-1 are closed during the post accident period in accordance with plant procedures and consistent with requirements of the related protection, safeguards, or essential service systems.

The opening angle of the containment vent isolation valves is being limited as an analysis demonstrates valve operability against accident containment pressures provided the valves are limited to a maximum opening angle of 60°. The containment purge supply and exhaust isolation valves are required to be closed during plant operation above cold shutdown.

REFERENCES


(1) FSAR - Section 14.3.6
(2) FSAR - Appendix 5A, Section 3.1.8
(3) FSAR - Section 5.1.1.1
(4) FSAR - Section 5.2






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L.1



TABLE 3.6-1

**NON-AUTOMATIC CONTAINMENT ISOLATION VALVES**

**OPEN CONTINUOUSLY OR INTERMITTENTLY FOR PLANT OPERATION**

| VALVE NO. | VALVE NO. | VALVE NO. |
|---|---|---|
| AC-MOV-744 | SI-MOV-1835B | SWN-71-4 |
| AC-MOV-1870 | SI-859A | SWN-71-5 |
| AC-MOV-743 | SI-859C | SA-24-1 |
| AC-MOV-822A | AC-752F | SA-24-2 |
| AC-MOV-822B | AC-753F | PS-PCV-1111-1 |
| SP-990C | AC-752J | PS-PCV-1111-2 |
| AC-732 | AC-753J | SP-MOV-990A |
| SI-MOV-885A | SWN-41-1 | SP-MOV-990B |
| SI-MOV-885B | SWN-43-1 | SI-1814A |
| SI-MOV-888A | SWN-41-2 | SI-1814B |
| SI-MOV-888B | SWN-43-2 | SI-1814C |
| CH-MOV-205 | SWN-41-3 | PS-7 |
| CH-MOV-226 | SWN-43-3 | PS-8 |
| CH-227 | SWN-41-4 | PS-9 |
| CH-MOV-250A | SWN-43-4 | PS-10 |
| CH-MOV-441 | SWN-41-5 | SP-SOV-506[1] |
| CH-MOV-250B | SWN-43-5 | SP-SOV-507[1] |
| CH-MOV-442 | SWN-44-1 | SP-SOV-508[1] |
| CH-MOV-250C | SWN-51-1 | SP-SOV-509[1] |
| CH-MOV-443 | SWN-44-2 | SP-SOV-510[1] |
| CH-MOV-250D | SWN-51-2 | SP-SOV-511[1] |
| CH-MOV-444 | SWN-44-3 | SP-SOV-512[1] |
| SI-869A | SWN-51-3 | SP-SOV-513[1] |
| SI-869B | SWN-44-4 | SP-SOV-514[1] |
| SI-878A | SWN-51-4 | SP-SOV-515[1] |
| SI-878B | SWN-44-5 | SP-SOV-516[1] |
| SI-MOV-851A | SWN-51-5 | CB-3 |
| SI-MOV-850A | SWN-71-1 | CB-4 |
| SI-MOV-850C | SWN-71-2 | CB-7 |
| SI-MOV-1835A | SWN-71-3 | CB-8 |

[1] Note: These series valves have non-redundant Phase A automatic signals and therefore are treated as non-automatic containment isolation valves.

Amendment No. ~~86, 102, 105, 115, 152~~

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Add SR 3.6.3.5



M.4

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS

SEE CTS MASTER MARKUP

| | Check | Frequency |
|---|---|---|
| 1. Control Rods | Rod drop times of all control rods | 24M |
| 2. Control Rods | Movement of at least 10 steps in any one direction of all control rods | Every 31 days during reactor critical operations |
| 3. Pressurizer Safety Valves | Set Point | 24M* |
| 4. Main Steam Safety Valves | Set Point | 24M |
| 5. Containment Isolation System (SR 3.6.3.6) | Automatic actuation | 24M } SR 3.6.3.6 |
| 6. Refueling System Interlocks | Functioning | Each refueling, prior to movement of core components |
| 7. Primary System Leakage | Evaluate | 5 days/week |
| 8. Diesel Generators Nos. 31, 32 & 33 Fuel Supply | Fuel Inventory | Weekly |
| 9. Turbine Steam Stop And Control Valves | Closure | Not to exceed 6 months** |
| 10. L.P. Steam Dump System (6 lines) | Closure | Monthly |
| 11. Service Water System | Each pump starts and operates for 15 minu... (unless already operating) | Monthly |
| 12. City Water Connections to Charging Pumps and Boric Acid Piping | Temporary connections available and valves operable | 24M |

SEE CTS MASTER MARKUP

* ~~Pressurizer Safety Valve setpoint test due no later than May 1996 may be deferred until the next refueling outage but no later than May 31, 1997.~~ TSCR 97-156

** The turbine steam stop and control valves shall be tested at a frequency determined by the methodology presented in WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency," as updated by Westinghouse Report, WOG-TVTF-93-17, "Update of BB-95/96 Turbine Valve Failure Rates and Effect on Destructive Overspeed Probabilities." The maximum test interval for these valves shall not exceed six months. Surveillance interval extension as per Technical Specification 1.12 is not applicable to the maximum test interval.



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SEE ITS 5.15, Cont Leak Rate Test Prog

LA.1

E. Containment Isolation Valves

SR 3.6.3.8 ~~1.~~ Verify the combined leakage rate for all containment bypass leakage paths, Table 4.4-1 lists required isolation valves, is ≤ 0.6La when pressurized ≥ Pa in accordance with the Containment Leakage Rate Testing Program.

SEE ITS 3.6.9

~~2.~~ Verify the leakage rate of water from the Isolation Valve Seal Water System is ≤ 14,700 cc/hr when pressurized ≥ 1.1 Pa, in accordance with the Containment Leakage Rate Testing Program.

~~3.~~ Verify the leakage rate of water into the containment from isolation valves sealed with the service water system is ≤ 0.36 gpm per fan cooler unit when pressurized ≥ 1.1 Pa, in accordance with the Containment Leakage Rate Testing Program.

Add SR 3.6.3.3 and Note

Add SR 3.6.3.4 and Note

M.3

Add SR 3.6.3.5

M.4

F. DELETED

G. DELETED

H. DELETED

A-1

Basis

The containment is designed for a pressure of 47 psig. [1] While the reactor is operating, the internal environment of the containment will be air at essentially atmospheric pressure and an average maximum temperature of approximately 130°F. The Design Basis Accidents (DBA) that represent a challenge to the containment structure are a Loss of Coolant Accident (LOCA) and a Main Steam Line Break (MSLB). The limiting calculated peak containment pressure of 42.40 psig is a result of the MSLB [7], which is less than the stated design pressure of 47 psig. In addition, DBA analyses demonstrate that the calculated peak containment temperature will remain less than the Equipment Qualification (EQ) envelope temperature of 290 degrees F.

The containment structure is designed to contain, within established leakage limits, radioactive material that may be released from the reactor core following a DBA. The containment was designed with an allowable leakage rate of 0.1 weight percent of containment air per day. This leakage rate, used to evaluate offsite doses resulting from DBAs is defined in 10CFR 50 Appendix B as $L_a$; the maximum allowable containment leakage rate at the calculated peak containment internal pressure ($P_a$) resulting from the limiting DBA. The allowable leakage rate represented by $L_a$ forms the basis for the acceptance criteria imposed on all containment leakage rate testing performed in accordance with the program required by Technical Specification 6.14. The minimum test pressure for this program, based on the current value of $P_a$ is 42.40 psig. Analyses which established the previous minimum test pressure of 42.42 psig were performed to support an increase of the ultimate heat sink temperature. [4] The conclusions of that analysis regarding heat sink temperature, as incorporated by Technical Specification Amendment 98, remain valid.

Prior to initial operation, the containment was strength-tested at 54 psig and was leak-tested. The acceptance criterion for this pre-operational leakage rate test was established as 0.075 W/o (.75 $L_a$) per 24 hours at 40.6 psig and 263°F, which were the peak accident pressure and temperature conditions at that time. This leakage rate is consistent with the construction of the containment, [2] which is equipped with a Weld Channel and Penetration Pressurization System for continuously pressurizing the containment penetrations and the channels over certain containment liner welds. These channels were independently leak-tested during construction.

The safety analysis has been performed on the basis of a leakage rate of 0.10 W/o per day for 24 hours. With this leakage rate and with minimum containment engineered safeguards operating, the public exposure would be well below 10CFR100 values in the event of the design basis accident. [3]

Maintaining the containment operable requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in surveillance requirement 4.4.D does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage



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A.1



prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test is required to be <0.6 $L_a$ for combined Type B and C leakage, and < 0.75 $L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 $L_a$. At ≤ 1.0 $L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. Surveillance requirement frequencies are as required by the Containment Leakage Rate Testing Program. Thus, Specification 1.12 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

The Weld Channel and Containment Penetration Pressurization System (WCCPPS)[5] is in service continuously to monitor leakage from potential leak paths such as the containment personnel lock seals and weld channels, containment penetrations, containment liner weld channels, double-gasketed seals and spaces between certain containment isolation valves and personnel door locks. A leak would be expected to build up slowly and would, therefore, be noted before design limits are exceeded. Remedial action can be taken before the limit is reached. The sensitive leakage rate test of the WCCPPS demonstrates that pressurized containment penetrations and liner inner weld seams are within a leakage acceptance criteria that will allow the air receivers and the standby source of gas pressure, nitrogen cylinders, to provide a 24 hour supply of gas to the system. The WCCPPS is not credited for limiting containment isolation valve leakage and the sensitivity test is not used for demonstrating compliance with containment isolation valve leakage criteria. The frequency of the sensitive leakage test reflects an extension of 25 percent from the 24 month refueling cycle and, therefore, Specification 1.12 (which allows Frequency extensions) does not apply[10].

Maintaining containment air locks operable requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. The surveillance requirement reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program. Thus, Specification 1.12 (which allows Frequency extensions) does not apply. During normal plant operation, containment personnel lock door seals are continuously pressurized after each closure by the WCCPPS. Whenever containment integrity is required, verification is made that seals repressurize properly upon closure of an air lock door. The verification meets the intent of the 10 CFR 50 Appendix J requirements.[8]





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The containment isolation valve surveillance requirement ensures that the combined leakage rate of all containment bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves, and, when pressurizing between valves, the total leakage of all the valves being tested) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. The Frequency is required by the Containment Leakage Rate Testing Program. This surveillance requirement simply imposes additional acceptance criteria. The service water lines to the containment fan cooler units and the lines supplied water by the Isolation Valve Seal Water System (IVSWS)[6] have containment isolation valves that are hydrostatically tested. Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of offsite doses are met. The Frequency is required by the Containment Leakage Rate Testing Program. Sufficient water is available in the Isolation Valve Seal Water System, Primary Water System, Service Water System, Residual Heat Removal System, and the City Water System to assure a sealing function for at least 30 days. The leakage limit for the Isolation Valve Seal Water System is consistent with the design capacity of the Isolation Valve Seal Water supply tank. The seal water provided by these systems is credited with limiting containment leakage (the measured leakage is not considered part of the allowable containment leakage).

The 350 psig test pressure, achieved either by normal Residual Heat Removal System operation or hydrostatic testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the hydrostatic test pressure for the containment sump return line of 100 psig gives an adequate margin over the highest pressure within the line after a design basis accident. A recirculation system leakage of 2 gal./hr. will limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident.

The maximum permissible inleakage rate from the containment isolation valves sealed with service water for the full 12-month period of post accident recirculation without flooding the internal recirculation pumps is 0.36 gpm per fan cooler.

REFERENCES


(1) FSAR - Section 5
(2) FSAR - Section 5.1.7
(3) FSAR - 14.3.5
(4) WCAP - 12269 Rev. 1, "Containment Margin Improvement Analysis for IP-3 Unit 3"
(5) FSAR - Section 6.6
(6) FSAR - Section 6.5
(7) Nuclear Safety Evaluation 98-3-013-MULT, "Integrated Safety Evaluation of 24-Month Cycle Instrument Channel Uncertainties," Revision 0, dated March 3, 1998.
(8) SECL-96-103, Indian Point Unit 3 Safety Evaluation of 24-Month Fuel Cycle Phase I Instrument Channel Uncertainties, June 1996
(9) Indian Point 3 Safety Evaluation Report, Supplement 2, December 1975.
(10) NRC Safety Evaluation Related to Amendment 129 to Operating License DPR-64.


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LA.1

TABLE 4.4-1 (Page 1 of 7)

CONTAINMENT ISOLATION VALVES

| Valve No. | Penetration Number(1) | Test Fluid(2) | Minimum Test Pressure (PSIG)(8) |
|---|---|---|---|
| RC-AOV-549 | 1 | Water(4) | 47 |
| RC-AOV-548 | 1 | Water(4) | 47 |
| RC-518 | 2 | Gas | 43 |
| RC-AOV-550 | 2 | Gas | 43 |
| RC-AOV-552 | 3 | Water(4) | 47 |
| RC-AOV-519 | 3 | Water(4) | 47 |
| AC-741 | 4 | Water(5) | 47(3) |
| AC-MOV-744 | 4 | Nitrogen(4) | 43(3) |
| SI-MOV-888A | 5 | Nitrogen(4) | 43 |
| SI-MOV-888B | 5 | Nitrogen(4) | 43 |
| AC-AOV-958 | 5 | Nitrogen(4) | 43 |
| SP-AOV-959 | 5 | Nitrogen(4) | 43 |
| SP-990C | 5 | Nitrogen(4) | 43 |
| AC-MOV-1870 | 5 | Nitrogen(4) | 43 |
| AC-MOV-743 | 5 | Nitrogen(4) | 43 |
| AC-732 | 6 | Nitrogen(4) | 43(3) |
| SI-MOV-885A | 7 | Water(5) | 47 |
| SI-MOV-885B | 7 | Water(5) | 47 |
| CH-AOV-201 | 8 | Water(4) | 47 |
| CH-AOV-202 | 8 | Water(4) | 47 |
| CH-MOV-205 | 9 | Water(4) | 47 |
| CH-MOV-226 | 9 | Water(4) | 47 |
| CH-227 | 9 | Water(4) | 47 |
| CH-MOV-250A | 10 | Water(4) | 47 |
| CH-MOV-441 | 10 | Water(4) | 47 |
| CH-MOV-250B | 10 | Water(4) | 47 |
| CH-MOV-442 | 10 | Water(4) | 47 |
| CH-MOV-250C | 10 | Water(4) | 47 |

LA.1

TABLE 4.4-1 (Page 2 of 7)

CONTAINMENT ISOLATION VALVES

| Valve No. | Penetration Number(1) | Test Fluid(2) | Minimum Test Pressure(PSIG)(3) |
|---|---|---|---|
| CH-MOV-443 | 10 | Water(4) | 47 |
| CH-MOV-250D | 10 | Water(4) | 47 |
| CH-MOV-444 | 10 | Water(4) | 47 |
| CH-MOV-222 | 11 | Water(4) | 47 |
| SP-AOV-956E | 12 | Water(4) | 47 |
| SP-AOV-956F | 12 | Water(4) | 47 |
| SI-869A | 14 | Water(4) | 47 |
| SI-867A | 14 | Gas | 43 |
| SI-878A | 14 | Gas | 43 |
| SI-869B | 14 | Water(4) | 47 |
| SI-867B | 14 | Gas | 43 |
| SI-878B | 14 | Gas | 43 |
| SI-MOV-1835A | 15 | Nitrogen(4) | 43 |
| SI-MOV-1835B | 15 | Nitrogen(4) | 43 |
| SI-MOV-851A | 15 | Water(4) | 47 |
| SI-MOV-850A | 15 | Water(4) | 47 |
| SI-MOV-850C | 15 | Water(4) | 47 |
| SI-859A | 16 | Water(4) | 47 |
| SI-859C | 16 | Water(4) | 47 |
| NNE-1610 | 17 | Gas | 43 |
| NNE-AOV-863 | 17 | Gas | 43 |
| SP-AOV-956G | 18 | Water(4) | 47 |
| SP-AOV-956H | 18 | Water(4) | 47 |
| WD-AOV-1786 | 19 | Water(4) | 47 |
| WD-AOV-1787 | 19 | Water(4) | 47 |

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TABLE 4.4-1 (Page 3 of 7)

CONTAINMENT ISOLATION VALVES

| Valve No. | Penetration Number (1) | Test Fluid(2) | Minimum Test Pressure(PSIG)(8) |
|---|---|---|---|
| WD-AOV-1610 | 19 | Gas | 43 |
| WD-1616 | 19 | Gas | 43 |
| WD-AOV-1788 | 20 | Water(4) | 47 |
| WD-AOV-1789 | 20 | Water(4) | 47 |
| WD-AOV-1702 | 21 | Water(4) | 47 |
| WD-AOV-1705 | 21 | Water(4) | 47 |
| AC-MOV-797 | 22 | Water(4) | 47 |
| AC-MOV-769 | 22 | Water(4) | 47 |
| AC-MOV-784 | 23 | Water(4) | 47 |
| AC-MOV-786 | 23 | Water(4) | 47 |
| AC-FCV-625 | 24 | Water(4) | 47 |
| AC-MOV-789 | 24 | Water(4) | 47 |
| AC-AOV-791 | 29 | Water(4) | 47 |
| AC-AOV-798 | 29 | Water(4) | 47 |
| AC-AOV-796 | 30 | Water(4) | 47 |
| AC-AOV-793 | 30 | Water(4) | 47 |
| WD-AOV-1728 | 31 | Water(4) | 47 |
| WD-AOV-1723 | 31 | Water(4) | 47 |
| VS-PCV-1234 | 32 | Gas(7) | 43 |
| VS-PCV-1235 | 32 | Gas(7) | 43 |
| VS-PCV-1236 | 33 | Gas(7) | 43 |
| VS-PCV-1237 | 33 | Gas(7) | 43 |
| CA-PCV-1229 | 34 | Gas(7) | 43 |
| CA-PCV-1230 | 34 | Gas(7) | 43 |
| BD-PCV-1215 | 37 | Water(4) | 47 |
| BD-PCV-1215A | 37 | Water(4) | 47 |
| BD-PCV-1214 | 37 | Water(4) | 47 |
| BD-PCV-1214A | 37 | Water(4) | 47 |
| BD-PCV-1216 | 37 | Water(4) | 47 |
| BD-PCV-1216A | 37 | Water(4) | 47 |
| BD-PCV-1217 | 37 | Water(4) | 47 |
| BD-PCV-1217A | 37 | Water(4) | 47 |

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TABLE 4.4-1 (Page 4 of 7)

CONTAINMENT ISOLATION VALVES

| Valve No. | Penetration Number(1) | Test Fluid(2) | Minimum Test Pressure(PSIG)(8) |
|---|---|---|---|
| BD-PCV-1224 | 38 | Water(4) | 47 |
| BD-PCV-1224A | 38 | Water(4) | 47 |
| BD-PCV-1223 | 38 | Water(4) | 47 |
| BD-PCV-1223A | 38 | Water(4) | 47 |
| BD-PCV-1225 | 38 | Water(4) | 47 |
| BD-PCV-1225A | 38 | Water(4) | 47 |
| BD-PCV-1226 | 38 | Water(4) | 47 |
| BD-PCV-1226A | 38 | Water(4) | 47 |
| SWN-41-1 | 39 | Water(6) | 47 |
| SWN-43-1 | 39 | Water(6) | 47 |
| SWN-42-1 | 39 | Water(6) | 47 |
| SWN-41-2 | 39 | Water(6) | 47 |
| SWN-43-2 | 39 | Water(6) | 47 |
| SWN-42-2 | 39 | Water(6) | 47 |
| SWN-41-3 | 39 | Water(6) | 47 |
| SWN-43-3 | 39 | Water(6) | 47 |
| SWN-42-3 | 39 | Water(6) | 47 |
| SWN-41-4 | 39 | Water(6) | 47 |
| SWN-43-4 | 39 | Water(6) | 47 |
| SWN-42-4 | 39 | Water(6) | 47 |
| SWN-41-5 | 39 | Water(6) | 47 |
| SWN-43-5 | 39 | Water(6) | 47 |
| SWN-42-5 | 39 | Water(6) | 47 |
| SWN-44-1 | 40 | Water(6) | 47 |
| SWN-51-1 | 40 | Water(6) | 47 |
| SWN-44-2 | 40 | Water(6) | 47 |
| SWN-51-2 | 40 | Water(6) | 47 |
| SWN-44-3 | 40 | Water(6) | 47 |
| SWN-51-3 | 40 | Water(6) | 47 |
| SWN-44-4 | 40 | Water(6) | 47 |
| SWN-51-4 | 40 | Water(6) | 47 |

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TABLE 4.4-1 (Page 5 of 7)

CONTAINMENT ISOLATION VALVES

| Valve No. | Penetration Number(1) | Test Fluid(2) | Minimum Test Pressure (PSIG)(8) |
|---|---|---|---|
| SWN-44-5 | 40 | Water(6) | 47 |
| SWN-51-5 | 40 | Water(6) | 47 |
| SWN-71-1 | 40 | Water(6) | 47 |
| SWN-71-2 | 40 | Water(6) | 47 |
| SWN-71-3 | 40 | Water(6) | 47 |
| SWN-71-4 | 40 | Water(6) | 47 |
| SWN-71-5 | 40 | Water(6) | 47 |
| SA-24-1 | 41 | Water(4) | 47 |
| SA-24-2 | 41 | Water(4) | 47 |
| VS-FCV-1170 | 48 | Gas(7) | 43 |
| VS-FCV-1171 | 48 | Gas(7) | 43 |
| VS-FCV-1172 | 49 | Gas(7) | 43 |
| VS-FCV-1173 | 49 | Gas(7) | 43 |
| VS-FCV-1190 | 50 | Gas(7) | 43 |
| VS-FCV-1191 | 50 | Gas(7) | 43 |
| VS-FCV-1192 | 50 | Gas(7) | 43 |
| SP-MOV-990A | 51 | Nitrogen(4) | 43 |
| SP-MOV-990B | 51 | Nitrogen(4) | 43 |
| SP-AOV-956A | 52 | Water(4) | 47 |
| SP-AOV-956B | 52 | Water(4) | 47 |
| SP-AOV-956C | 53 | Water(4) | 47 |
| SP-AOV-956D | 53 | Water(4) | 47 |
| SI-1814A | 54 | Gas | 43 |
| SI-1814B | 55 | Gas | 43 |
| SI-1814C | 56 | Gas | 43 |
| SP-SOV-506 | 57 | Gas(7) | 43 |
| SP-SOV-507 | 57 | Gas(7) | 43 |



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LA.1

**TABLE 4.4-1** (Page 6 of 7)

| CONTAINMENT ISOLATION VALVES | | | |
|---|---|---|---|
| Valve No. | Penetration Number[1] | Test Fluid[2] | Minimum Test Pressure (PSIG)[8] |
| SP-SOV-508 | 57 | Gas[7] | 43 |
| SP-SOV-509 | 57 | Gas[7] | 43 |
| SP-SOV-510 | 57 | Gas[7] | 43 |
| SP-SOV-511 | 57 | Gas[7] | 43 |
| SP-SOV-512 | 57 | Gas[7] | 43 |
| SP-SOV-513 | 57 | Gas[7] | 43 |
| SP-SOV-514 | 57 | Gas[7] | 43 |
| SP-SOV-515 | 57 | Gas[7] | 43 |
| SP-SOV-516 | 57 | Gas[7] | 43 |
| IA-39 | 64 | Gas | 43 |
| IA-PCV-1228 | 64 | Gas | 43 |
| PS-7 | 65 | Gas[7] | 43 |
| PS-10 | 65 | Gas[7] | 43 |
| PS-8 | 65 | Gas[7] | 43 |
| PS-9 | 65 | Gas[7] | 43 |
| CB-1 | 69 | Gas | 43 |
| CB-2 | 69 | Gas | 43 |
| CB-3 | 69 | Gas[7] | 43 |
| CB-4 | 69 | Gas[7] | 43 |
| CB-5 | 68 | Gas | 43 |
| CB-6 | 68 | Gas | 43 |
| CB-7 | 68 | Gas[7] | 43 |
| CB-8 | 68 | Gas[7] | 43 |
| DW-AOV-1 | 70 | Water[4] | 47 |
| DW-AOV-2 | 70 | Water[4] | 47 |

LA.1

TABLE 4.4-1 (Page 7 of 7)

CONTAINMENT ISOLATION VALVES

NOTES:

1. Reference: FSAR Table 5.2-1, Penetration No.
2. Gas Test Fluid indicates either nitrogen or air as test medium.
3. Testable only when at cold shutdown.
4. Isolation Valve Seal Water System.
5. Sealed by Residual Heat Removal System fluid.
6. Sealed by Service Water System.
7. Sealed by Weld Channel and Penetration Pressurization System.
8. The minimum test pressure may be reduced by 2 psig until the current requirements associated with the Boron Injection Tank are removed (see Tech Spec 3.3.A.3.b).

Add SR 3.6.3.2 M.2

ITS 3.6.3

A.1

Pressure Relief A.11

4.13 Containment Vent and Purge System

**Applicability**

Mode 1, 2, 3 and 4 A4

This specification applies to the surveillance requirements of the containment vent and purge system during normal operations and when reactor fuel is anticipated to be moved before the reactor has been subcritical for at least 421* hours.

SEE ITS 3.9.3

**Objective**

To verify the operability of the containment vent and purge system. A.2

**Specification**

The following surveillance shall be performed as stated.

A. Isolation Valves

Mode 1,2,3,4 A.4

SR 3.6.3.1 1. Each month verify that the containment purge supply and exhaust isolation valves are closed during ~~operation above cold shutdown~~.

Pressure Relief A.11

SR 3.6.3.7 2. At least once per 24 months verify ~~that the mechanical~~ stops on the containment vent isolation valve (PCV-1190, -1191, -1192) actuator is limited to the valve opening angle to 60° (90° = full open).

B. HEPA Filters and Charcoal Absorbers

SEE ITS 5.5.10

If fuel movement is to take place before the reactor has been subcritical for at least 421* hours, the containment vent and purge system shall be demonstrated operable as follows:

1. Within 18 months prior to fuel movement and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank within 18 months prior to fuel movement, or (2) after structural maintenance on the HEPA filter or charcoal adsorber housing within 18 months prior to fuel movement, which could effect system operation:

   a. Verify that the charcoal adsorbers remove ≥ 99% of halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the ventilation system at the operating flow ± 10%.

   b. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place while operating the ventilation system at the operating flow rate ± 10%.

2. Within 18 months prior to fuel movement and after every 720 hours of system operation, subject a representative sample of carbon from the charcoal adsorbers to a laboratory analysis and verify within 31 days a removal efficiency of ≥ 90% for radioactive methyl iodine at an operating air flow velocity ± 20% per test 5.b in Table 2 of Regulatory Guide 1.52, March 1978.

SEE ITS 3.9.3

* Movement of irradiated VANTAGE + fuel assemblies before the reactor has been subcritical for ≥550 hours requires operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal adsorbers.



Amendment No. ~~30~~, ~~62~~, ~~128~~, ~~131~~, 175

Basis

The containment purge supply and exhaust isolation valves are required to be closed during plant operation above cold shutdown. Containment purge supply or exhaust isolation valve closure may be verified by way of the position indication lights, the weld channel and penetration pressurization system or visual means. The maximum opening angle of the containment vent isolation valves is being limited as an analysis demonstrates valve operability against accident containment pressures provided the valves are limited to an opening angle of 60°.

A.1

The operability of the HEPA filter and charcoal absorber system and the resulting iodine removal capacity are consistent with accident analyses. The representative carbon sample will be two inches in diameter with a length equal to the thickness of the bed.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.3:
# "Containment Isolation Valves"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.6.3 - Containment Isolation Valves

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.6.A.1 specifies that containment integrity (as defined in CTS 1.10) shall not be violated; and, CTS 1.10.1 through CTS 1.10.3 and CTS 1.10.6 specify that nonautomatic containment isolation valves not required to be open during accident conditions must be closed and blind flanges installed where required and automatic containment isolation valves are either operable or in the closed position or isolated by a closed manual valve or flange. ITS LCO 3.6.3 maintains the identical requirements by specifying that each containment isolation valve must be Operable with Operability described in the Bases. In conjunction with this change, the CTS definition of Containment Integrity is deleted because it contains information that is more appropriately contained in the LCOs (and SRs) which establish the requirements for containment integrity and the Bases associated with these LCOs and SRs. This change is needed to improve clarity and ensure requirements are fully understood and consistently applied. This reorganization of requirements is an administrative change with no impact on safety because the ITS requirements are reasonable interpretations of the existing requirements except for the specific changes identified and justified in the discussion of changes for each LCO addressing containment issues.

A.4 CTS 3.6.A.1, CTS 3.6.A.3, CTS 3.6.D, and CTS 4.13 specify the Applicability for containment isolation valves as whenever the reactor is above cold shutdown. ITS 3.6.3 maintains this Applicability by requiring that containment isolation valves are Operable in Modes 1, 2, 3 and 4 (i.e., above cold shutdown). This is an administrative change with no impact on safety because there is no change to the CTS Applicability.

A.5 ITS LCO 3.6.3, Actions Note 4, requires entry into applicable Conditions and Required Actions of LCO 3.6.1, Containment, when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria. Additionally, ITS LCO 3.6.3, Actions Note 5, requires entry into applicable Conditions and Required Actions of LCO 3.6.9, Isolation Valve Seal Water (IVSW) System, when required IVSW supply to a penetration flowpath is isolated. These Notes are needed to eliminate any ambiguity about the governing LCO and associated Conditions and Required Actions for the situations described. This change is needed because ITS divides containment Operability requirements into four separate LCOs: ITS 3.6.1 which requires that the primary containment is Operable; ITS 3.6.2 which requires that the primary containment air locks are Operable; ITS 3.6.3 which requires that each containment isolation valve is Operable; and, ITS LCO 3.6.9 which requires that IVSW is Operable. This change adds notes intended to ensure that the appropriate LCOs are recognized for any Condition and that appropriate Required Actions are implemented. This reorganization of requirements is an administrative change with no impact on safety except for the specific changes identified and justified in the discussion of changes for each LCO addressing containment issues.

A.6 ITS LCO 3.6.3, Actions Note 2, specifies that separate Condition entry is allowed for each penetration flow path. This allowance provides explicit recognition that the ITS is designed to allow completely separate re-entry into any Condition for each penetration flow path (but not for individual isolation valves). This includes separate tracking of Completion Times based on this re-entry. This allowance is consistent with an unstated assumption in the CTS. Therefore, the addition of this Note is an administrative change with no impact on safety.

A.7 ITS LCO 3.6.3, Actions Note 3, requires entry into applicable Conditions and Required Actions for systems made inoperable by containment isolation valves. This Note is added to eliminate ambiguity concerning the applicability of ITS LCO 3.0.6 when containment isolation valves render another system inoperable. ITS LCO 3.0.6 specifies that the Conditions and Required Actions associated with a supported system are not required to be entered when a supported system LCO is not met. Only the support system LCO Actions are required to be entered. Without the addition of ITS LCO 3.6.3, Actions Note 3, ambiguity could exist as to the need to enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves that must be closed to satisfy containment isolation requirements. This is an administrative change with no impact on safety because it is a reasonable interpretation of the existing requirement and is consistent with current practice.

A.8 ITS 3.6.3, Conditions A, B and C, are preceded by Notes identifying the containment isolation valve configuration for which the Condition is applicable. Specifically, Condition A and B are only applicable to those penetration flow paths with two or more containment isolation valves when one (Condition A) or both (Condition B) valves are inoperable. Although most penetrations have two containment isolation valves, the term "two or more" is used so that Conditions A and B apply to the pressure relief line penetration which has three valves in series. Condition C provides the appropriate actions for penetration flow paths with only one containment isolation valve and a closed system. The addition of these Notes does not eliminate any existing requirements or establish any new requirements and the Notes are intended to provide direction for the proper use of the LCO. This reorganization of requirements is an administrative change with no impact on safety except for the specific changes identified and justified in the discussion of changes for each ITS LCO 3.6.3 Conditions and Required Action.

A.9 CTS 1.10.6 specifies that the compensatory action for an inoperable containment isolation valve includes isolating a penetration flow path with a closed manual valve or flange that meets the same design criteria as the isolation valve. ITS LCO 3.6.3, Required Actions, allow use of one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured to be used in lieu of an inoperable containment isolation valve. Although the ITS is more specific in defining what constitutes a closed manual valve, the intent of both the CTS and ITS is that the penetration flow path is isolated using at least one isolation barrier that cannot be adversely affected by a single active failure. Therefore, this is an administrative change with no impact on safety because the ITS requirements are consistent with a reasonable interpretation of the existing requirements.

A.10 In accordance with 10 CFR 50, Appendix A, Criterion 56, containment integrity requires one automatic isolation valve inside and one automatic isolation valve outside containment with the option of allowing one closed isolation valve for either or both of the automatic isolation valves such that the design maintains the ability to tolerate a single failure. Criterion 56 also allows other configurations that are acceptable on some other defined basis (e.g., the IP3 design of both isolation valves outside containment). Additionally, ITS LCO 3.6.3 recognizes that one automatic isolation valve and the associated closed system inside containment constitute a single failure tolerant containment isolation boundary (see 3.6.3, DOC L.2). This is acceptable because the closed system acts as a highly reliable penetration isolation boundary. This is an administrative change with no impact on safety because it is explicit recognition in Technical Specifications of a containment isolation configuration used in the IP3 design. Additionally, use of one automatic isolation valve and the associated closed system to constitute a single failure tolerant containment isolation boundary is consistent with industry practice.

A.11 CTS 3.6.D and 4.13 use the term containment vent to describe the containment penetration that includes pressure relief isolation valves PCV-1190, PCV-1191, and PCV-1192 and which is used to handle the normal pressure changes in the Containment during reactor power operation. FSAR 5.3.2.5 and control room labeling identify this system as the Containment Pressure Relief Line. ITS will use the term Containment Pressure Relief Line for this system to be consistent with FSAR 5.3.2.5 and control room labeling. This is an administrative change with no adverse impact on safety because there is no change to the existing requirements.

MORE RESTRICTIVE

M.1 CTS 3.6.A.3 and CTS 3.6.D, which apply to inoperable containment isolation valves in accordance with CTS 1.10, require that containment integrity is restored within one hour whenever containment integrity requirements are not met. In accordance with CTS 1.10.6, containment integrity is restored if at least one manual valve or flange is used to isolate the penetration flow path (see 3.6.2, DOC A.9). ITS LCO 3.6.3, Required Actions A.1, B.1 and C.1, maintain this requirement (see 3. 6.2, DOC L.2); however, Required Actions A.2 and C.2, add the additional requirement to verify the affected penetration flow path is isolated once per 31 days for isolation devices outside containment. Additionally, for penetrations with isolation devices inside containment, this verification is also required prior to entering Mode 4 from Mode 5 if not performed within the previous 92 days. In either case, isolation devices in high radiation areas may be verified by use of administrative means.

The change is needed to ensure periodic verification that penetration flow paths no longer capable of being automatically isolated remain isolated by an acceptable substitute. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering Mode 4 from Mode 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility. Additionally, the new requirement for periodic verification of manual valves used to substitute for inoperable containment isolation valves do not apply to isolation devices located in high radiation areas because the probability of misalignment of these devices is small once they have been verified to be in the proper position.

This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification that isolation devices used to substitute for inoperable containment isolation valves remain in the correct position. Therefore, this change has no adverse impact on safety.

M.2 CTS 3.6.A and CTS 4.13 do not include any limits on the amount of time or the reasons that containment pressure relief (See ITS 3.6.3, DOC A.11) isolation valves (PCV-1190, PCV-1191, and PCV-1192) may be open. This is acceptable because both CTS and ITS limit the opening angle of the containment pressure relief isolation valves to an angle at which analysis indicates the valve will operate against containment accident pressures. However, ITS SR 3.6.3.2 adds an additional restriction that containment pressure relief isolation valves may be opened only as necessary for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. ITS SR 3.6.3.2 requires verification of this status every 31 days. This change is needed because it ensures that containment pressure relief line valves are opened only as necessary to satisfy their intended function. This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification that containment pressure relief isolation valves are opened only as necessary to satisfy their intended function. Therefore, this change has no adverse impact on safety.

M.3 CTS 1.10.2 and 1.10.3 require, as a condition of containment integrity, that blind flanges are installed and test connections, vents and drains are capped or closed with blind flanges; however, CTS 3.6.A and CTS 4.13 do not include any requirements for the periodic verification that manual isolation valves and blind flanges are positioned or installed as required. ITS SR 3.6.3.3 and ITS SR 3.6.3.4 are added to require periodic verification that isolation valves and blind flanges not locked, sealed or otherwise secured are positioned or installed as required.

ITS SR 3.6.3.3, governing valves and flanges outside containment, has a required Frequency of once per 31 days. This Frequency is acceptable because these devices are operated under administrative controls and the probability of their misalignment is low.

ITS SR 3.6.3.4, governing valves and flanges inside containment, has a required Frequency of prior to entering Mode 4 from Mode 5 if not performed within the previous 92 days. This Frequency is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility. Isolation devices in high radiation areas, both inside and outside containment, may be verified by use of administrative means because the restricted access to these areas provides a high degree of assurance that the valves will not be mispositioned inadvertently.

The new requirement for periodic verification of manual valves does not apply to isolation devices that are locked, sealed or otherwise secured in position because these devices are positioned in accordance with plant administrative programs and the probability of misalignment of these devices is small once they have been verified to be in the proper position.

This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification that isolation devices remain in the correct position. Therefore, this change has no adverse impact on safety.

M.4 CTS 3.6.A and CTS 4.13 do not include any requirements for the periodic verification that the isolation time of each power operated automatic containment isolation valve is within limits. ITS SR 3.6.3.5 is added to require periodic verification that each automatic containment isolation valve is within limits at a Frequency in accordance with the Inservice Testing Program. This change is needed to provide periodic verification that the containment isolation time is less than or equal to that assumed in the safety analyses. This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification that isolation devices function within the limits assumed in the safety analyses. Therefore, this change has no adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 1.10.1 and CTS 3.6.A.1 allow nonautomatic containment isolation valves identified in Table 3.6-1 to be opened if necessary for plant operation but only as long as necessary to perform the intended function. ITS 3.6.3, Actions Note 1, expands this allowance to include any containment isolation valve (except for the 36 inch containment purge supply and exhaust valves) and will include manual valves used to substitute for inoperable automatic isolation valves. In conjunction with this change, ITS 3.6.3, Actions Note 1, stipulates that valves opened using this allowance must be subject to administrative controls consisting of stationing at the valve controls a dedicated operator in continuous communication with the control room. This change, expanding the allowance to intermittently open any containment isolation valve under administrative controls and deleting CTS Table 3.6-1, is acceptable because the allowance is used infrequently, the valves will be opened only as long as necessary to perform a required function, and there is a low probability of an event requiring containment isolation. Additionally, the administrative controls imposed by ITS LCO 3.6.3 are sufficient to ensure that any penetration can be rapidly isolated when a need for containment isolation is indicated. Therefore, this change has no significant impact on safety.

L.2 CTS 3.6.A.3 and CTS 3.6.D, which apply to inoperable containment isolation valves in accordance with CTS 1.10, require that containment integrity is restored within one hour whenever containment integrity requirements are not met. In accordance with 10 CFR 50, Appendix A, Criterion 56, containment integrity requires two automatic isolation valves with the option of allowing one closed isolation valve for either or both of the automatic isolation valves such that the design maintains the ability to tolerate a single failure. Therefore, when one or both containment isolation valves in a penetration flow path are not Operable, CTS 3.6.A.3 and CTS 3.6.D require that a closed manual valve or equivalent be substituted for the inoperable valve within one hour if containment integrity is lost.

ITS LCO 3.6.3 maintains the same requirement but differentiates between loss of single failure tolerance and a loss of function in the determination of an acceptable out of service time (AOT). Specifically, ITS LCO 3.6.3, Required Action A.1, extends the AOT from one hour to 4 hours for loss of single failure tolerance; ITS LCO 3.6.3, Required Action B.1, maintains the AOT at one hour for loss of function; and, ITS LCO 3.6.3, Required Action C.1, extends the AOT from one hour to 72 hours for penetration flow paths with only one containment isolation valve but protecting a closed system.

This change is acceptable for the following reasons: a) for penetrations with two automatic isolation valves, automatic isolation of the penetration will still occur with only one Operable automatic isolation valve; b) for penetrations associated with closed systems and one automatic isolation valve, the closed system acts as a highly reliable penetration isolation boundary with minimal need for redundancy provided by the automatic isolation valve; c) in both cases, the time without single failure tolerance is limited to 4 hours; and, d) there is a low probability of an event requiring containment isolation during the AOT. Therefore, this change has no significant impact on safety.

REMOVED DETAIL

LA.1 CTS 3.6.A.1, CTS 1.10, and CTS 3.6.D require that containment isolation valves are Operable and specifies that the isolation valves are identified in CTS Table 4.4-1, Containment Isolation Valves. ITS 3.6.3 requires that the same valves are Operable; however, the lists identifying individual containment isolation valves will be relocated to the FSAR. This is acceptable because containment isolation valves are clearly defined by their function and IP3 design documents; therefore, not listing the valves in the Technical Specifications does not affect the Technical Specification requirement that the containment isolation valves are Operable.

Maintaining the lists of containment isolation valves in the FSAR is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, are designed to assure that changes to the FSAR do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight is maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.3:
# "Containment Isolation Valves"

## PART 4:

## No Significant Hazards Considerations
## for
## Changes between CTS and ITS
## that are
## Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.6.3 - Containment Isolation Valves

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change allows any containment isolation valve (except purge supply and exhaust valves) not just non automatic valves to be opened without taking the Required Actions for breach of containment. However, valves opened using this allowance must be subject to administrative controls consisting of stationing a dedicated operator in continuous communication with the control room at the valve controls. This change will not result in a significant increase in the probability of an accident previously evaluated because the position of containment isolation valves has no effect on the initiators of any accident. This change will not result in a significant increase in the consequences of an accident previously evaluated because the administrative controls imposed by ITS LCO 3.6.3 are sufficient to ensure that any penetration can be rapidly isolated when a need for containment isolation is indicated. Additionally, the allowance is used infrequently, the valves will be opened only as long as necessary to perform a required function, and the low probability of an event requiring containment isolation.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the administrative controls imposed by ITS LCO 3.6.3 are sufficient to ensure that any penetration can be rapidly isolated when a need for containment isolation is indicated.

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LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change differentiates between those failures of containment isolation valves that result in a breach of containment and those failures that result only in a loss of isolation redundancy. This change then extends that allowable out of service time (AOT) for a loss of redundancy from 1 hour to 4 hours and extends the AOT from one hour to 72 hours for penetration flow paths with only one containment isolation valve but protecting a closed system.

   This change will not result in a significant increase in the probability of an accident previously evaluated, nor result in a significant increase in the consequences of an accident previously evaluated because of the following reasons: a) for penetrations with two automatic isolation valves, automatic isolation of the penetration will still occur with only one Operable automatic isolation valve; b) for penetrations associated with closed systems and one automatic isolation valve, the closed system acts as a highly reliable penetration isolation boundary with minimal need for redundancy provided by the automatic isolation valve; c) in both cases, the time without single failure tolerance is limited to 72 hours; and, d) there is a low probability of an event requiring containment isolation during the limited AOT.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because of the following reasons: a) for penetrations with two automatic isolation valves, automatic isolation of the penetration will still occur with only one Operable automatic isolation valve; b) for penetrations associated with closed systems and one automatic isolation valve, the closed system acts as a highly reliable penetration isolation boundary with minimal need for redundancy provided by the automatic isolation valve; c) in both cases, the time without single failure tolerance is limited to 72 hours; and, d) there is a low probability of an event requiring containment isolation during the limited AOT.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.6.3:
# "Containment Isolation Valves"

# PART 5:

# NUREG-1431
# Annotated to show differences between NUREG-1431 and ITS

## Status of NUREG 1431 Generic Changes for ITS 3.6.3

This ITS Specification is based on NUREG-1431 Specification No. **3.6.3**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| BWROG-017 | **051** | REVISE CONTAINMENT REQUIREMENTS DURING HANDLING IRRADIATED FUEL AND CORE ALTERATIONS (REQUIREMENTS LIMITED TO "RECENTLY" IRRADIATED FUEL) | NRC Review | Not Incorporated | **N/A** |
| CEOG-062 | **145 R0** | ADD ACTION TO VERIFY FLOW PATH IS ISOLATED WHEN 2 CIVS INOPERABLE | See Next Rev | | **N/A** |
| CEOG-062 R1 | **145 R1** | ADD ACTION TO VERIFY FLOW PATH IS ISOLATED WHEN 2 CIVS INOPERABLE | TSTF to Rewrite | | **N/A** |
| CEOG-112 | **196 R0** | REVISE ISOLATION DEVICES TO INCLUDE ASME/ANSI EQUIVALENT METHODS | Rejected by NRC | Not Incorporated. OI to evaluate if benefit. | **N/A** |

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.3:
## "Containment Isolation Valves"

| | | | | | |
|---|---|---|---|---|---|
| WOG-034 | **030 R0** | EXTEND THE COMPLETION TIME FOR INOPERABLE ISOLATION TO A CLOSED SYSTEM TO 72 HOURS | See Next Rev | See Rev 1 | **N/A** |
| WOG-034 R1 | **030 R1** | EXTEND THE COMPLETION TIME FOR INOPERABLE ISOLATION TO A CLOSED SYSTEM TO 72 HOURS | See Next Rev | Not Incorporated. Evaluate when Rev 2 issued. | **N/A** |
| WOG-038 | **044** | ADD A NOTE TO THE CONTAINMENT ISOLATION VALVE LCO WHICH EXEMPTS MSSVS, MSIVS, MFIV, MFRVS, AND ADVS NRC REJECTS: TSTF ACCEPTS | Rejected by NRC | Not Incorporated | **N/A** |
| WOG-039 | **045 R0** | EXEMPT VERIFICATION OF CIVS THAT ARE LOCKED, SEALED OR OTHERWISE SECURED | See Next Rev. | See Rev. 1. | **N/A** |
| WOG-039 R1 | **045 R1** | EXEMPT VERIFICATION OF CIVS THAT ARE LOCKED, SEALED OR OTHERWISE SECURED | Approved by NRC | Incorporated | **T.1** |
| WOG-040 | **046** | CLARIFY THE CIV SURVEILLANCE TO APPLY ONLY TO AUTOMATIC ISOLATION VALVES | See Next Rev. | See Next Rev. | **N/A** |
| WOG-040 R1 | **046 R1** | CLARIFY THE CIV SURVEILLANCE TO APPLY ONLY TO AUTOMATIC ISOLATION VALVES | Approved by NRC | Incorporated | **T.2** |
| WOG-042 | **052** | IMPLEMENT 10 CFR 50, APPENDIX J, OPTION B | TSTF to Rewrite | TSTF is CLB | **N/A** |
| WOG-091 | | ALLOW ADMINISTRATIVE MEANS OF POSITION VERIFICATION FOR LOCKED OR SEALED VALVES COMPONENTS ARE MODIFIED TO ALLOW THE VERIFICATION TO BE "BY ADMINISTRATIVE MEANS". | TSTF Review | Not Incorporated | **N/A** |

## 3.6 CONTAINMENT SYSTEMS

<CTS>

### 3.6.3 Containment Isolation Valves ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~

<3.6.A.1>
<1.10.6>
<DOC A.3>

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

<3.6.A.1>
<DOC A.4>

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

Insert: 3.6-8-01

<1.10.1><DOC L.1> <3.6.A.1>

NOTES

1. Penetration flow path(s) ~~[~~except for ~~[42]~~ 36 inch purge valve flow paths~~]~~ may be unisolated intermittently under administrative controls.

<DOC A.6>

2. Separate Condition entry is allowed for each penetration flow path.

<DOC A.7>

3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.

<DOC A.5>

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

<DOC A.8>
<1.10.6>
<DOC L.2>
<DOC A.9>

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. ---------NOTE--------- Only applicable to penetration flow paths with ~~two~~ or more containment isolation valves. --------------------- One or more penetration flow paths with one containment isolation valve inoperable ~~[except for purge valve or shield building bypass leakage not within limit]~~. | A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. <br> AND | 4 hours |

(continued)

INSERT: 3.6-8-01

5. Enter applicable Conditions and Required Actions of LCO 3.6.9, "Isolation Valve Seal Water (IVSW) System," when required IVSW supply to a penetration flowpath is inoperable.

<DOC A.5>

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| <DOC H.1><br>A. (continued) | A.2 ---------NOTE---------<br>Isolation devices in high radiation areas may be verified by use of administrative means.<br>---------------------<br>Verify the affected penetration flow path is isolated. | Once per 31 days for isolation devices outside containment<br>**AND**<br>Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment |
| <DOC A.8><br>B. ---------NOTE---------<br>Only applicable to penetration flow paths with two or more containment isolation valves.<br>----------------------<br><1.10.6> <3.6.A.3> <DOC A.9><br>One or more penetration flow paths with two containment isolation valves inoperable ~~[except for purge valve or shield building bypass leakage not within limit]~~. | B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. | 1 hour |

(continued)

ACTIONS (continued)

<DOC A.8>
<1.10.6>
<DOC L.2>

<3.6.A3>
<DOC H.1>
<Doc A.9>
<DOC A.10>

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| C. ---------NOTE---------<br>Only applicable to penetration flow paths with only one containment isolation valve and a closed system.<br>---------------------<br><br>One or more penetration flow paths with one containment isolation valve inoperable. | C.1 | Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. | [~~4~~] hours 72 |
| | **AND** | | |
| | C.2 | --------NOTE---------<br>Isolation devices in high radiation areas may be verified by use of administrative means.<br>--------------------<br><br>Verify the affected penetration flow path is isolated. | Once per 31 days |
| D. Shield building bypass leakage not within limit. | D.1 | Restore leakage within limit. | 4 hours |
| E. One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits. | E.1 | Isolate the affected penetration flow path by use of at least one [closed and de-activated automatic valve, closed manual valve, or blind flange]. | 24 hours |
| | **AND** | | |

(continued)

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| E. (continued) | E.2 | --------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. --------------------- | |
| | | Verify the affected penetration flow path is isolated. | Once per 31 days for isolation devices outside containment<br>**AND**<br>Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment |
| | **AND** | | |
| | E.3 | Perform SR 3.6.3.7 for the resilient seal purge valves closed to comply with Required Action E.1. | Once per [92] days |
| <3.6.A.3> <3.6.D> D ~~F~~. Required Action and associated Completion Time not met. | D ~~F~~.1 | Be in MODE 3. | 6 hours |
| | **AND** | | |
| | D ~~F~~.2 | Be in MODE 5. | 36 hours |

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | | FREQUENCY |
|---|---|---|---|
| ⟨4.13.A.1⟩ ⟨3.6.D⟩ | SR 3.6.3.1 Insert: 3.6-12-01 | Verify each ~~[42]~~ 36 inch ~~purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition E of this LCO.~~ | 31 days |
| ⟨DOC M.2⟩ | SR 3.6.3.2 Insert: 3.6-12-02 | Verify each [8] 10 inch ~~purge valve is closed, except when the [8] inch containment purge~~ valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. | 31 days |
| ⟨DOC M.3⟩ ⟨1.10.2⟩ ⟨1.10.3⟩ | SR 3.6.3.3 | -------------------NOTE------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ------------------------------------------ <br><br> Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls. | 31 days |

(continued)

T.1

NUREG-1431 Markup Inserts
ITS SECTION 3.6.3 - Containment Isolation Valves

<u>INSERT: 3.6-12-01</u>

purge supply and exhaust isolation valve is sealed closed.

<u>INSERT: 3.6-12-02</u>

pressure relief isolation valve is closed, except when these

SURVEILLANCE REQUIREMENTS (continued)

<CTS>

| | | SURVEILLANCE | FREQUENCY | |
|---|---|---|---|---|
| | SR 3.6.3.4 | --------NOTE-------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------------- | | |
| <DOC M.3> <1.10.2> <1.10.3> | | Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls. | Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days | T.1 |
| <DOC M.4> | SR 3.6.3.5 | Verify the isolation time of ~~each power operated and~~ each automatic power operated containment isolation valve is within limits. | ~~[~~ In accordance with the Inservice Testing Program ~~or 92 days~~ ~~]~~ | T.2 |
| | ~~SR 3.6.3.6~~ | ~~Cycle each weight or spring loaded check valve testable during operation through one complete cycle of full travel, and verify each check valve remains closed when the differential pressure in the direction of flow is ≤ [1.2] psid and opens when the differential pressure in the direction of flow is ≥ [1.2] psid and < [5.0] psid.~~ | ~~92 days~~ | |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| ~~SR 3.6.3.7~~ | ~~Perform leakage rate testing for containment purge valves with resilient seals.~~ | ~~184 days~~<br>~~AND~~<br>~~Within 92 days after opening the valve~~ |
| ⟨Table 4.1-3, Item 5⟩ SR 3.6.3.~~8~~ 6 | Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal. | ~~[18]~~ 24 months |
| ~~SR 3.6.3.9~~ | ~~Cycle each weight or spring loaded check valve not testable during operation through one complete cycle of full travel, and verify each check valve remains closed when the differential pressure in the direction of flow is ≤ [1.2] psid and opens when the differential pressure in the direction of flow is ≥ [1.2] psid and ≤ [5.0] psid.~~ | ~~18 months~~ |
| ⟨4.13.A.2⟩ ⟨3.6.D⟩ ⟨DOC A.11⟩ SR 3.6.3.~~10~~ 7 | Verify each ~~[ ]~~ 10 inch containment purge valve ~~is blocked to restrict the valve from opening > [50]%~~ Insert: 3.6-14-01 | ~~[18]~~ 24 months |

(continued)

**INSERT 3.6-14-01:**

containment pressure relief line isolation valve is blocked to restrict valve opening to ≤ 60 degrees.

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| ~~SR 3.6.3.11~~ | ~~Verify the combined leakage rate for all shield building bypass leakage paths is ≤ [ $L_a$] when pressurized to ≥ [ psig].~~ | ~~-----NOTE----- SR 3.0.2 is not applicable~~<br>~~In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions~~ |

Insert:
3.6-15-01

NUREG-1431 Markup Inserts
ITS SECTION 3.6.2 - Containment Isolation Valves

<CTS>

INSERT 3.6-15-01:

| | | |
|---|---|---|
| SR 3.6.3.8 <4.4.E.1> | Verify the combined leakage rate for all containment bypass leakage paths in accordance with the Containment Leakage Rate Testing Program. | In accordance with the Containment Leakage Rate Testing Program |

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, ~~the purge and exhaust valves receive an isolation signal on a containment high radiation condition.~~ As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

Insert: B 3.6-29-01

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the

(continued)

INSERT: B 3.6-29-01

the Containment Purge supply and exhaust isolation valves (FCV-1170, FCV-1171, FCV-1172, and FCV-1173) and the containment pressure relief isolation valves (PCV-1190, PCV-1191, and PCV-1192) close when high radiation levels are detected by the Containment Air Particulate Monitor (R-11) or Containment Radioactive Gas Monitor (R-12). Containment purge and containment pressure relief are also isolated when high radiation levels are detected in the plant vent.

BASES

BACKGROUND (continued)

time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

Containment

36

Shutdown Purge System ([42] inch purge valves)

Insert: B3.6-30-01

The Shutdown Purge System operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size, the [42] inch purge valves ~~in some units~~ are not qualified for automatic closure from their open position under DBA conditions. Therefore, the [42] inch purge valves are ~~normally~~ maintained closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

36 sealed must be

Containment Pressure Relief Line

10

Minipurge System ([8] inch purge valves)

Insert: B 3.6-30-02

The Minipurge System operates to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access, and

b. Equalize internal and external pressures.

Since the valves used in the Minipurge System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4.

Insert: B 3.6-30-03

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.6.3 - Containment Isolation Valves

<u>INSERT: B 3.6-30-01</u>

, consisting of purge supply and exhaust isolation valves FCV-1170, FCV-1171, FCV-1172, and FCV-1173,

<u>INSERT: B 3.6-30-02</u>

consisting of pressure relief isolation valves PCV-1190, PCV-1191, and PCV-1192,

<u>INSERT: B 3.6-30-03</u>

Containment pressure relief line isolation valve opening is limited by mechanical stops so that opening angle is limited to an angle at which analysis indicates the valve will operate against containment accident pressures. Additionally, pressure relief isolation valve opening must be limited to the time necessary for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.

The containment pressure relief line is isolated in during CORE ALTERATIONS and during movement of irradiated fuel inside containment in accordance with requirements established in LCO 3.9.3, Containment Penetrations.

BASES

A

APPLICABLE SAFETY ANALYSES (continued)

is

The ~~DBAs~~ that ~~result~~ in a release of radioactive material within containment ~~are~~ a loss of coolant accident (LOCA) ~~and a rod ejection accident~~ (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves ~~(including containment purge valves)~~ are minimized. The safety analyses assume that the ~~[42]~~ 36 inch purge valves are closed at event initiation. sealed

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, $L_a$. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

~~[The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are provided with diverse power sources, motor operated and pneumatically operated spring closed, respectively. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.]~~

Insert: B 3.6-31-01

~~[~~The ~~purge valves~~ may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment purge valves due to failure in the control circuit associated with each valve. Again, the purge system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.~~]~~

Insert: B 3.6-31-02

The containment isolation valves satisfy Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36

NUREG-1431 Markup Inserts
ITS SECTION 3.6.3 - Containment Isolation Valves

INSERT: B 3.6-31-01

Containment Purge supply and exhaust isolation valves (FCV-1170, FCV-1171, FCV-1172, and FCV-1173)

INSERT: B 3.6-31-02

The term sealed closed, as applied to containment isolation valves, is not intended to describe leak tightness. Sealed closed isolation valves must be under administrative controls that assure the valve cannot be inadvertently opened. Administrative controls includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator (Ref. 3). Sealed closed barriers include blind flanges and sealed closed isolation valves including closed manual valves, closed remote-manual valves, and closed automatic valves which remain closed after a loss-of-coolant accident. Sealed closed barriers may be used in place of any automatic isolation valve.

BASES (continued)

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

36

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The [42] inch purge valves must be maintained sealed closed ~~[or have blocks installed to prevent full opening]. [Blocked purge valves also actuate on an automatic signal.]~~ The valves covered by this LCO are listed ~~along with their associated stroke times~~ in the FSAR (Ref. 2).

H

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. ~~These passive isolation valves/devices are those listed in Reference 1.~~

(Ref. 3)

~~Purge valves with resilient seals [and secondary containment bypass valves] must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.~~

Insert: B36-32-01

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, ~~"Containment Penetrations."~~

3

(continued)

**INSERT: B 3.6-32-01**

Manually operated containment isolation valves on essential lines that are required to be open, at least for a time, during post accident conditions are OPERABLE if they can be closed in accordance with design assumptions. Essential lines are those lines required to mitigate an accident, or which, if unavailable, could increase the magnitude of the event. Also, those lines which, if available, would be used in the short term (24 to 36 hours) to restore the plant to normal operation following an event which has resulted in containment isolation (Ref. 4).

BASES (continued)

ACTIONS

Insert: B3.6-33-01

36

The ACTIONS are modified by a Note allowing penetration flow paths, except for [42] inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust ~~directl~~y from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls. ~~A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.~~

2

~~A second~~ Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

3

The ACTIONS are further modified by ~~a third~~ Note, which ensures appropriate remedial actions are taken, ~~if necessary,~~ if the affected systems are rendered inoperable by an inoperable containment isolation valve.

Containment isolation valve

In the event ~~the air lock~~ leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

Insert: B3.6-33-02

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable ~~[except for purge valve or shield building bypass leakage not within limit]~~, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.6.3 - Containment Isolation Valves

INSERT: B 3.6-33-01

Note 1, which allows penetration flow paths that are isolated in accordance with Required Actions,

INSERT: B 3.6-33-02

The ACTIONS are further modified by Note 5, which ensures appropriate remedial actions are taken if required IVSW supply to a penetration flowpath is inoperable. Specifically, Note 5 directs entry into the applicable Conditions and Required Actions of LCO 3.6.9.

BASES

ACTIONS

A.1 and A.2 (continued)

failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured (Ref. 3). For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. ~~Rather, it~~ This Action involves verification, through a system walkdown, that ~~those~~ isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment (e.g., one of the three containment pressure relief isolation valves), the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two or more containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Insert: B 3.6-34-01

(continued)

INSERT: B 3.6-34-01

Although most penetrations have two containment isolation valves, the term "two or more" is used so that Condition A includes penetrations such as the containment pressure relief line which has three valves in series.

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

B.1

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Insert B3.6-35-01

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

NUREG-1431 Markup Inserts
ITS SECTION 3.6.3 - Containment Isolation Valves

INSERT: B 3.6-35-01

Although most penetrations have two containment isolation valves, the term "two or more" is used so that Condition B includes penetrations such as the containment pressure relief line which has three valves in series.

BASES

ACTIONS (continued)

C.1 and C.2

, other than the closed system.

(Ref. 3)

72

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the [#] hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by a Note that applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

(continued)

BASES

ACTIONS (continued)

D.1

With the shield building bypass leakage rate not within limit, the assumptions of the safety analyses are not met. Therefore, the leakage must be restored to within limit within 4 hours. Restoration can be accomplished by isolating the penetration(s) that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration(s) and the relative importance of secondary containment bypass leakage to the overall containment function.

E.1, E.2, and E.3

In the event one or more containment purge valves in one or more penetration flow paths are not within the purge valve leakage limits, purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a [closed and de-activated automatic valve, closed manual valve, or blind flange]. A purge valve with resilient seals utilized to satisfy Required Action E.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.7. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action E.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the containment purge valve with resilient seal that is isolated in accordance with Required Action E.1, SR 3.6.3.7 must be performed at least once every [92] days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.7, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 3). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per [92] days was chosen and has been shown to be acceptable based on operating experience.

~~F~~D.1 and ~~F~~D.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1 36 Insert B 3.6-39-01

Each 42 inch containment purge valve is required to be verified sealed closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. ~~A containment purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness.~~ Insert: B 3.6-39-02 The Frequency is a result of an NRC initiative, Generic Issue B-24 (Ref. 4), 5 91 related to containment purge valve use during plant operations. ~~In the event purge valve leakage requires entry into Condition E, the Surveillance permits opening one purge valve in a penetration flow path to perform repairs.~~

SR 3.6.3.2

Insert: B 3.6-39-03

This SR ensures that the minipurge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the minipurge valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The minipurge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

Containment pressure relief line isolation

As long as valve opening angle is limited in accordance with SR 3.~~6.3.7~~.

(continued)

# NUREG-1431 Markup Inserts
# ITS SECTION 3.6.3 - Containment Isolation Valves

**INSERT: B 3.6-39-01**

purge supply and exhaust isolation valve (FCV-1170, FCV-1171, FCV-1172, and FCV-1173)

**INSERT: B 3.6-39-02**

The term sealed closed, as applied to containment isolation valves, is not intended to describe leak tightness. Sealed closed isolation valves must be under administrative controls that assure the valve cannot be inadvertently opened. Administrative controls includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator (Ref. 3). Sealed closed barriers include blind flanges and sealed closed isolation valves including closed manual valves, closed remote-manual valves, and closed automatic valves which remain closed after a loss-of-coolant accident.

**INSERT: B 3.6-39-03**

(PCV-1190, PCV-1191, and PCV-1192)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.3

(T.1)

Insert: B 3.6-40-01

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open.

Insert B 3.6-40-02

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.4

Insert B 3.6-40-01

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.6.2 - Containment Isolation Valves

INSERT: B 3.6-40-01

and not locked, sealed, or otherwise secured

INSERT: B 3.6-40-02

This SR does not apply to valves that are locked, sealed or otherwise secured in the closed position because these valves were verified to be in the correct position when locked, sealed or otherwise secured.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.4 (continued)

administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open.

Insert: B3.6-41-01

T.1

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.5

T.2

Verifying that the power operated the isolation time of each ~~power operated and~~ automatic containment isolation valve is within limits as specified in the FSAR is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. [The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program ~~or 92 days~~.]

~~SR 3.6.3.6~~

~~In subatmospheric containments, the check valves that serve a containment isolation function are weight or spring loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.6 requires verification of the operation of the check valves that are testable during unit operation. The Frequency of 92 days is consistent with the Inservice Testing Program requirement for valve testing on a 92 day Frequency.~~

(continued)

INSERT: B 3.6-41-01

This SR does not apply to valves that are locked, sealed or otherwise secured in the closed position because these valves were verified to be in the correct position when locked, sealed or otherwise secured.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.6.3.7~~

~~For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 3).~~

~~Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.~~

SR 3.6.3.~~8~~6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The
24 [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be 24 acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.6.3.9~~

~~In subatmospheric containments, the check valves that serve a containment isolation function are weight or spring loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.9 verifies the operation of the check valves that are not testable during unit operation. The Frequency of 18 months is based on such factors as the inaccessibility of these valves, the fact that the unit must be shut down to perform the tests, and the successful results of the tests on an 18 month basis during past unit operation.~~

SR 3.6.3.10 7

~~Reviewer's Note: This SR is only required for those units with resilient seal purge valves allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices on the valves that are not permanently installed.~~

Insert: B3.6-43-01

Verifying that each ~~[42] inch containment purge valve is blocked to restrict opening to ≤ [50]%~~ is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. ~~At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open.~~ The 18 month Frequency is appropriate because the blocking devices are typically not removed ~~only during a refueling outage~~.

pressure relief line

24

SR 3.6.3.11 8

This SR ensures that the combined leakage rate of all ~~shield building bypass~~ leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the

Containment

Insert: B3.6-43-02

(continued)

INSERT: B 3.6-43-01

containment pressure relief line isolation valve, PCV-1190, PCV-1191, and PCV-1192, is blocked to restrict valve opening to ≤ 60 degrees. This verification

INSERT: B 3.6-43-02

that are not sealed by the Isolation Valve Seal Water System or sealed by the RHR system or sealed by the service water system.

BASES

8

SURVEILLANCE REQUIREMENTS

SR 3.6.3.~~11~~ (continued)

maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. ~~This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with Appendix J). The Frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions (and therefore, the Frequency extensions of SR 3.0.2 may not be applied), since the testing is an Appendix J, Type C test. This SR simply imposes additional acceptance criteria.~~

Insert. B 3.6-44-01

~~[By pass leakage is considered part of $L_a$. [Reviewer's Note: Unless specifically exempted].]~~

REFERENCES

1. FSAR, Section ~~[15]~~. 14

2. FSAR, Section ~~[6.2]~~. 6

~~3. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."~~

Insert: B 3.6-44-02

5 ~~4~~. Generic Issue B-24.

NUREG-1431 Markup Inserts
ITS SECTION 3.6.3 - Containment Isolation Valves

INSERT: B 3.6-44-01

This testing is performed in accordance with the requirements, Frequency and acceptance criteria established in Specification 5.5.15, Containment Leakage Rate Testing Program. This program was established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by IP3 specific approved exemptions. This program conforms to guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995." In the event containment isolation valve leakage results in exceeding the overall containment leakage rate, entry into the applicable Conditions and Required Actions of LCO 3.6.1 is required.

INSERT: B 3.6-44-02

3. Standard Review Plan Section 6.2.4, Item II.3.f.

4. FSAR, Section 5.2.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.3: "Containment Isolation Valves"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.6.3 - Containment Isolation Valves

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 This change maintains IP3 current licensing basis related to the use of 10 CFR 50, Appendix J, Option B, for containment leak rate testing which was approved on June 17, 1997 as part of Amendment 174. This change is based on Generic Change TSTF-52 (WOG-42), Revision 0, which is currently being reviewed by the NRC.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-45 (WOG-39), Rev 1, which revises SR 3.6.3.3 and SR 3.6.6.3 to specify that only containment isolation valves that are not locked, sealed, or otherwise secured are required to be verified closed. This change is acceptable because it makes containment isolation valve requirements consistent with requirements in ECCS (SR 3.5.2.2), AFW (SR 3.7.5.1.), and SW (SR 3.7.9.1).

T.2 This change incorporates Generic Change TSTF-46 (WOG-40), Rev 1, which revises SR 3.6.3.5 to delete reference to verifying the isolation time of "each power operated" containment isolation valve and only require verification of each "automatic isolation valve." This change is needed because the Bases for this SR state that the "isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis." There may be valves credited as containment isolation valves which are power operated (i.e., can be remotely operated) that do not receive a containment isolation signal (e.g., a GDC 57 penetration). These power operated valves do not have an isolation time as assumed in the accident analyses since they require operator action. Therefore, deleting reference to power operated isolation valve time testing reduces the potential for misinterpreting the requirements of this SR while maintaining the assumptions of the accident analysis.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.4:
## "Containment Pressure"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.6 CONTAINMENT SYSTEMS

### 3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -2.0 psig and ≤ +2.5 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Containment pressure not within limits. | A.1 Restore containment pressure to within limits. | 1 hour |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br><u>AND</u><br>B.2 Be in MODE 5. | 6 hours<br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.6.4.1 Verify containment pressure is within limits. | 12 hours |

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.4 Containment Pressure

BASES

### BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). The containment can withstand an internal vacuum of 3 psig. The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

### APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. Cycle specific analysis results indicate that the worst case peak containment pressure could result from either a loss of coolant accident or a steam line break inside containment (Ref. 1).

The initial pressure condition used in the containment analysis was +2.5 psig. This analysis concluded that the containment design pressure of 47 psig would not be exceeded for either a major loss-of-coolant accident or for a main steam line break accident. The containment analysis results are presented in Reference 1 and the current value for peak containment pressure is listed in Specification 5.5.15, "Containment Leakage Rate Testing Program."

BASES

APPLICABLE SAFETY ANALYSES (continued)

The containment was also designed for an external pressure load equivalent to -3.0 psig (i.e., the containment can withstand an internal vacuum of 3 psig). The -2.0 psig specified as the Limiting Condition for Operation is based on limits associated with motor cooling.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36.

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that motor heating concerns are addressed.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

BASES

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

BASES

REFERENCES

1. FSAR, Section 14.3
2. 10 CFR 50, Appendix K.
3. FSAR Section 3.1.8, Appendix 5A.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.4:
## "Containment Pressure"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.6-1 | 86 TSCR 97-070 | 86 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| 3.6-3 | 8-30-95 TSCR 97-070 | 8-30-95 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |

3.6 CONTAINMENT SYSTEM

~~Applicability~~

~~Applies to the integrity of reactor containment.~~

~~Objective~~

~~To define the operating status of the reactor containment for plant operation.~~

~~Specification~~

A2

A. Containment Integrity

SEE ITS 3.6

1. Containment integrity (as defined in 1.10) shall not be violated unless the reactor is in the cold shutdown condition. Those valves to be opened continuously or intermittently are under administrative control and are open only as long as necessary to perform their intended function.

2. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is sufficient to maintain the shutdown margin equal to or greater than the requirements of specification 3.8.D.

SEE ITS 3.6

3. If the containment integrity requirements are not met when the reactor is above cold shutdown, containment integrity shall be restored within one hour or the reactor shall be in the hot shutdown condition within six hours and in cold shutdown condition within the next 30 hours.

B. Internal Pressure

LCO 3.6.4 If the internal pressure exceeds 2.5 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected within 1 hour or the ~~reactor shutdown~~ Mode 3 in 6 hours and Mode 5 in 36 hours

M.1

Applicability Mode 1, 2, 3 and 4

A.3



Amendment No. ~~34~~, ~~66~~

TSCR 97-070

Add SR 3.6.4.1

M.2

4.1

Regarding internal pressure limitations, the containment design pressure of 47 psig would not be exceeded for a major loss-of-coolant accident or for a main steam line break accident.[1] The loss-of-coolant accident event bounds the main steam line break accident from the containment peak pressures standpoint. The initial pressure condition used in the containment analysis was 2.5 psig.[1] The containment can withstand an internal vacuum of 3 psig.[2] The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

The requirement of a 50°F minimum containment ambient temperature is to assure that the minimum service metal temperature of the containment liner is well above the NDT + 30°F criterion for the liner material. [3]

Limiting maximum containment ambient temperature will ensure that the peak accident containment pressure does not exceed the design limit of 47 psig during steamline break or loss of coolant accidents. Environmentally and seismically qualified RTDs mounted on the crane wall above the containment fan cooler units inlet are normally used for measuring containment ambient temperature. Portable temperature sensing equipment may also be used, provided the criteria of 3.6.C.3 are met.

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function. [4] During periods of normal plant operations requiring containment integrity, some of the valves on this Table will be open either continuously or intermittently depending on requirements of the particular protection, safeguards or essential service systems. Those valves to be opened are under administrative control and are open only as long as necessary to perform their intended function. Some of the valves listed in Table 3.6-1 are closed during the post accident period in accordance with plant procedures and consistent with requirements of the related protection, safeguards, or essential service systems.

The opening angle of the containment vent isolation valves is being limited as an analysis demonstrates valve operability against accident containment pressures provided the valves are limited to a maximum opening angle of 60°. The containment purge supply and exhaust isolation valves are required to be closed during plant operation above cold shutdown.

REFERENCES

(1) FSAR - Section 14.3.6
(2) FSAR - Appendix 5A, Section 3.1.8
(3) FSAR - Section 5.1.1.1
(4) FSAR - Section 5.2



Amendment No. ~~62, 86, 98~~, ~~Revised by letter dated 8/30/95~~,

TSCR 97-070

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.4:
# "Containment Pressure"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

# DISCUSSION OF CHANGES
## ITS SECTION 3.6.4 - Containment Pressure

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.6.B requires that if the containment internal pressure exceeds specified limits, then the condition must be corrected or the reactor shutdown; however, no Completion Time is specified. Therefore, in accordance with CTS 3.0, the time is assumed to be zero and action is initiated without delay. Under the same conditions, ITS LCO 3.6.4 Required Action A.1, allows one hour for restoration of containment pressure before a reactor shutdown is required. This is an administrative change with no adverse impact on safety because it is a reasonable interpretation of the existing requirement

MORE RESTRICTIVE

M.1 CTS 3.6.B does not specify an Applicability for the limits on containment pressure; however, CTS 3.6.B establishes an implied Applicability of Modes 1 and 2 by requiring only that the reactor be shutdown if containment pressure limits are not met. ITS 3.6.4 maintains the requirement for the limits on containment pressure; however, ITS 3.6.4 expands the Applicability to Modes 1, 2, 3 and 4. This change is needed because a DBA could cause a release of radioactive material to containment if reactor coolant temperature is greater than 200°F. ITS LCO 3.6.4 is applicable in Modes 1, 2, 3 and 4 because containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained. This change has no adverse impact on safety.

M.2 CTS 3.6.B specifies that containment internal pressure must be maintained between +2.5 psig and -2.0 psig; however, there is no explicit requirement for periodic verification that this requirement is met. ITS SR 3.6.4.1 is added to verify every 12 hours that containment pressure is within required limits. This more restrictive requirement is acceptable because it ensures that unit operation remains within the limits assumed in the accident analysis. The 12 hour frequency was developed based on operating experience related to the trending of containment pressure variations during the applicable modes. The Frequency is considered adequate because of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition. This change has no adverse impact on safety.

DISCUSSION OF CHANGES
ITS SECTION 3.6.4 - Containment Pressure

LESS RESTRICTIVE

None

REMOVED DETAIL

None

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.4:
## "Containment Pressure"

## PART 4:

## No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.6.4 - Containment Pressure

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

There are no less restrictive changes for the adoption of this ITS.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.4:
## "Containment Pressure"

## PART 5:

## NUREG-1431
## Annotated to show differences between NUREG-1431 and ITS

**Status of NUREG 1431 Generic Changes for ITS 3.6.4**

This ITS Specification is based on NUREG-1431 Specification No. **3.6.4A**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| N/A | N/A | NO GENERIC CHANGES ARE POSTED AGAINST THIS SPECIFICATION. | Not Applicable | Not Applicable | N/A |

## 3.6 CONTAINMENT SYSTEMS

### 3.6.4~~A~~ Containment Pressure ~~(Atmospheric, Dual, and Ice Condenser)~~

⟨3.6.B⟩ LCO 3.6.4~~A~~ Containment pressure shall be ≥ [~~-0.3~~] -2.0 psig and ≤ [~~+1.5~~] +2.5 psig.

⟨DOC M.1⟩ APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| ⟨3.6.B⟩ ⟨DOC A.3⟩ | A. Containment pressure not within limits. | A.1 Restore containment pressure to within limits. | 1 hour |
| ⟨3.6.B⟩ ⟨DOC M.1⟩ | B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br><u>AND</u><br>B.2 Be in MODE 5. | 6 hours<br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| ⟨3.6.B⟩ ⟨DOC M.2⟩ | SR 3.6.4~~A~~.1 Verify containment pressure is within limits. | 12 hours |

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4~~A~~ Containment Pressure ~~(Atmospheric, Dual, and Ice Condenser)~~

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). ~~These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.~~

Insert: B3.6-45-04

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. ~~The worst case LOCA generates larger mass and energy release than the worst case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint~~ (Ref. 1).

Insert: B3.6-45-01

~~The initial pressure condition used in the containment analysis was [17.7] psia ([3.0] psig). This resulted in a maximum peak pressure from a LOCA of [53.9] psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, $P_a$, results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA, [44.1] psig, does not exceed the containment design pressure, [55] psig.~~

Insert: B 3.6-45-02

The containment was also designed for an external pressure load equivalent to ~~[-2.5]~~ -3.0 psig. ~~The inadvertent actuation of the Containment Spray System was analyzed to determine~~

Insert: B 3.6-45-03

(continued)

<u>INSERT: B 3.6-45-01</u>

Cycle specific analysis results indicate that the worst case peak containment pressure could result from either a loss of coolant accident or a steam line break inside containment.

<u>INSERT: B 3.6-45-02</u>

The initial pressure condition used in the containment analysis was +2.5 psig. This analysis concluded that the containment design pressure of 47 psig would not be exceeded for either a major loss-of-coolant accident or for a main steam line break accident. The containment analysis results are presented in Reference 1 and the current value for peak containment pressure is listed in Specification 5.15, Containment Leakage Rate Testing Program.

<u>INSERT: B 3.6-45-03</u>

(i.e.. the containment can withstand an internal vacuum of 3 psig. The -2.0 psig specified as the Limiting Condition for Operation is based on limits associated with motor cooling.

<u>INSERT: B 3.6-45-04</u>

The containment can withstand an internal vacuum of 3 psig. The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~the resulting reduction in containment pressure. The initial pressure condition used in this analysis was [-0.3] psig. This resulted in a minimum pressure inside containment of [-2.0] psig, which is less than the design load.~~

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment pressure satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that ~~the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.~~

motor heating concerns are addressed.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

(continued)

BASES (continued)

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4~~A~~.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. FSAR, Section ~~6.2~~. 14.3
2. 10 CFR 50, Appendix K.
3. FSAR 3.1.8, appendix 5A

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.4:
# "Containment Pressure"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.6.4 - Containment Pressure

## RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

## PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

## PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

## DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

## DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.5:
## "Containment Air Temperature"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.6 CONTAINMENT SYSTEMS

### 3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 130°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Containment average air temperature not within limit. | A.1 Restore containment average air temperature to within limit. | 8 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br><u>AND</u><br>B.2 Be in MODE 5. | 6 hours<br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.6.5.1 Verify containment average air temperature is within limit. | 24 hours |

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.5 Containment Air Temperature

BASES

---

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Containment Spray and Cooling systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

---

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment

BASES

APPLICABLE SAFETY ANALYSES (continued)

pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of the Containment Spray System, Residual Heat Removal System, and Containment Cooling System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature may be either a LOCA or a SLB. The initial containment average air temperature is assumed in the design basis analyses. The maximum containment air temperature and the design temperature are specified in (Ref. 1.)

The temperature limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design temperature for only a few seconds during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA LOCA or SLB.

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure may be either a LOCA or a SLB. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36.

BASES

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere.

A representative measurement of containment air temperature requires an arithmetic average of temperatures measured at no fewer than 4 locations. Environmentally and seismically qualified RTDs mounted on the crane wall above the containment fan cooler units inlet are normally used for measuring containment ambient temperature. Portable temperature sensing equipment may also be used.

The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. FSAR, Section 14.3.
2. 10 CFR 50.49.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.6.5:
# "Containment Air Temperature"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.6-2 | 3-16-95 TSCR 97-175 | 3-16-95 TSCR 97-175 | IPN 97-175 | Changes to Bases Pages | |
| 3.6-3 | 8-30-95 TSCR 97-070 | 8-30-95 TSCR 97-070 | IPN 97-070 | Clarification of Containment Integrity | Incorporated |
| T 4.1-1(5) | 169 TSCR 98-043 | 169 TSCR 98-043 | IPN 98-043 | Instrument Channel Surveillance Intervals Extended to 24 Months | Incorporated |

C. Containment Temperature

LCO 3.6.5 & Applicability

~~1.~~ ~~The reactor shall not be taken above the cold shutdown condition unless the containment ambient temperature is greater than 50°F.~~ LA.2

2. Containment ambient temperature shall not exceed 130°F when the reactor is above the cold shutdown condition. If the temperature is greater than 130°F, reduce the temperature to within the limit within 8 hours, or be in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. A.3

Req Act A.1

Req Act B.1, B.2

SR 3.6.5.1

3. Containment ambient temperature ~~as specified in 3.6.C.1 and 3.6.C.2 shall be the arithmetic average of temperatures measured at no fewer than 4 locations~~ at least once per 24 hours. LA.1

D. Containment Vent and Purge System

SEE ITS 3.6.3

The reactor shall not be taken above the cold shutdown condition unless the containment vent isolation valves (PCV - 1190, - 1191, - 1192) are closed or limited to a maximum valve opening angle of 60° (90° - full open) by mechanical means.

The reactor shall not be taken above the cold shutdown condition unless the containment purge supply and exhaust isolation valves (FCV - 1170, - 1171, - 1172, - 1173) are closed.

If the above conditions cannot be met within one hour, the reactor shall be in the hot shutdown condition within six hours and in the cold shutdown condition within the next 30 hours.

~~Basis~~

~~The Reactor Coolant System must be in the cold shutdown condition in order to relax containment integrity. When the Reactor Coolant System is in the cold shutdown condition, the pressurizer may have an internal temperature above 200°F for purposes of drawing and maintaining a steam bubble, provided that the reactor has been subcritical for at least 24 hours. Operation in this manner ensures that, in case of an accidental RCS coolant release under cold shutdown conditions, the ensuing offsite radiation doses will be within the limits of 10 CFR 100.~~

~~The shutdown margins are selected on the type of activities that are being carried out. The shutdown margin requirement of specification 3.8.D when the vessel head bolts are less than fully tensioned precludes criticality during refueling. When the reactor head is not to be removed, the specified cold shutdown margin of 1% Δ k/k precludes criticality in any occurrence.~~

A.1

A.1

Regarding internal pressure limitations, the containment design pressure of 47 psig would not be exceeded for a major loss-of-coolant accident or for a main steam line break accident.[1] The loss-of-coolant accident event bounds the main steam line break accident from the containment peak pressures standpoint. The initial pressure condition used in the containment analysis was 2.5 psig.[1] The containment can withstand an internal vacuum of 3 psig.[2] The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

The requirement of a 50°F minimum containment ambient temperature is to assure that the minimum service metal temperature of the containment liner is well above the NDT + 30°F criterion for the liner material. [3]

Limiting maximum containment ambient temperature will ensure that the peak accident containment pressure does not exceed the design limit of 47 psig during steamline break or loss of coolant accidents. Environmentally and seismically qualified RTDs mounted on the crane wall above the containment fan cooler units inlet are normally used for measuring containment ambient temperature. Portable temperature sensing equipment may also be used, provided the criteria of 3.6.C.3 are met.

Table 3.6-1 lists non-automatic valves that are designated as part of the containment isolation function. [4] During periods of normal plant operations requiring containment integrity, some of the valves on this Table will be open either continuously or intermittently depending on requirements of the particular protection, safeguards or essential service systems. Those valves to be opened are under administrative control and are open only as long as necessary to perform their intended function. Some of the valves listed in Table 3.6-1 are closed during the post accident period in accordance with plant procedures and consistent with requirements of the related protection, safeguards, or essential service systems.

The opening angle of the containment vent isolation valves is being limited as an analysis demonstrates valve operability against accident containment pressures provided the valves are limited to a maximum opening angle of 60°. The containment purge supply and exhaust isolation valves are required to be closed during plant operation above cold shutdown.

**REFERENCES**

(1) FSAR - Section 14.3.6
(2) FSAR - Appendix 5A, Section 3.1.8
(3) FSAR - Section 5.1.1.1
(4) FSAR - Section 5.2



Amendment No. ~~62, 86, 98~~, ~~Revised by letter dated 8/30/95~~,

TSCR 97-070

**TABLE 4.1-1** (Sheet 5 of 6)

| | Channel Description | Check | Calibrate | Test | Remarks |
|---|---|---|---|---|---|
| SEE CTS MASTER MARKUP | 37. Core Exit Thermocouples | D | 24M | N.A. | |
| | 38. Overpressure Protection System (OPS) | D | 18M (1) | 18M | 1) Calibration frequency for OPS sensors (RCS pressure and temperature) is 24 months |
| | 39. Reactor Trip Breakers | N.A. | N.A. | TM(1) | 1) Independent operation of under-voltage and shunt trip attachments |
| | | | | 24M(2) | 2) Independent operation of under-voltage and shunt trip from Control Room manual push-button |
| | 40. Reactor Trip Bypass Breakers | N.A. | N.A. | (1) | 1) Manual shunt trip prior to each use |
| | | | | 24M(2) | 2) Independent operation of under-voltage and shunt trip from Control Room manual push-button |
| | | | | 24M(3) | 3) Automatic undervoltage trip |
| | 41. Reactor Vessel Level Indication System (RVLIS) | D | 24M | N.A. | |
| LCO 3.6.5 | 42. ~~Ambient Temperature Sensors Within the Containment Building~~ | ~~D~~ | ~~24M~~ | ~~N.A.~~ | LA3 |
| SEE CTS MASTER MARKUP | 43. River Water Temperature # (installed) | S | 18M | N.A. | 1) Check against installed instrumentation or another portable device |
| | 44. River Water Temperature # (portable) | S (1) | Q (2) | N.A. | 2) Calibrate within 30 days prior to use and quarterly thereafter |
| | 45. Steam Line Flow | S | 24M | Q | Engineered Safety Features circuits only |

ITS 3.6.5

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.5: "Containment Air Temperature"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

## DISCUSSION OF CHANGES
## ITS SECTION 3.6.5 - Containment Air Temperature

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.6.C.2 specifies the Applicability for containment temperature limits as whenever the reactor is above cold shutdown. ITS 3.6.5 maintains this Applicability by requiring that Containment temperature is within specified limits in Modes 1, 2, 3 and 4 (i.e., above the cold shutdown). This is an administrative change with no impact on safety because there is no change to the Applicability.

DISCUSSION OF CHANGES
ITS SECTION 3.6.5 - Containment Air Temperature

MORE RESTRICTIVE

None

LESS RESTRICTIVE

None

REMOVED DETAIL

LA.1 CTS 3.6.C.3 requires that containment ambient temperature be the arithmetic average of temperatures measured at no fewer than 4 locations, at least once per 24 hours. ITS SR 3.6.5.1 maintains the requirement to verify every 24 hours that containment temperature is within required limits. However, the implementation details regarding the number and location of temperature detectors and the requirement to use an arithmetic average to calculate the temperature are not included in the ITS and are relocated to the ITS SR 3.6.5.1 Bases.

This change is acceptable because the requirement to maintain containment air temperature within specified limits and the requirements for periodic verification of these limits is maintained in the Technical Specifications. The design information that a representative measurement of containment air temperature requires an arithmetic average of temperatures measured at no fewer than 4 locations can be maintained in the Bases because there is no exemption from the Technical Specification requirement that containment air temperature must be maintained within specified limits.

Additionally, the Technical Specification Bases are subject to change control in accordance with ITS 5.5.12, Bases Control Program. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. This change is a less restrictive administrative change with no impact on safety.

LA.2 CTS 3.6.C.1 specifies that the reactor shall not be taken above the cold shutdown condition unless the containment ambient temperature is greater than 50°F. This requirement is not included in ITS 3.6.5 and is relocated to the FSAR and plant procedures.

This change, which allows the requirement that containment ambient temperature is greater than 50°F to be maintained in the FSAR and implemented by procedures, is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, are designed to assure that changes to the FSAR do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

LA.3 CTS Table 4.1-1, Item 42, requires that the channel be checked daily and calibrated every 24 months. ITS SR 3.6.5.1 maintains the requirement to verify every 24 hours that containment temperature is within required limits. Although the ITS SR 3.6.5.1 Bases specifies that four temperature sensors must be used (See ITS 3.6.5, DOC LA.1), there is no requirement to use specific instruments to satisfy ITS SR 3.6.5.1 and, consequently, no explicit requirements for periodic calibration of these instrument. Therefore, requirements for periodic verification and calibration of the installed ambient temperature sensors within the containment building is relocated to the and is relocated to the FSAR and plant procedures.

This change is acceptable because meeting the ITS SR 3.6.5 SRs requires at least four containment building temperature sensors and that these instruments are calibrated. Therefore, maintaining the requirement in Technical Specifications that containment ambient temperature must be verified within required limits every 24 hours and maintaining requirements for operation and calibration of instruments required to perform these verification in the FSAR and plant procedures provides an adequate level of assurance that containment ambient temperature will be maintained within required limits.

This change is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, are designed to assure that changes to the FSAR do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 require periodic submittal of FSAR and Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

## Indian Point 3
## Improved Technical Specifications (ITS)
## Conversion Package

# Technical Specification 3.6.5:
# "Containment Air Temperature"

## PART 4:

## No Significant Hazards Considerations
## for
## Changes between CTS and ITS
## that are
## Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.6.5 - Containment Air Temperature

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

There are no less restrictive changes for the adoption of this ITS.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.5:
## "Containment Air Temperature"

## PART 5:

## NUREG-1431
## Annotated to show differences between NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.6.5

This ITS Specification is based on NUREG-1431 Specification No. **3.6.5A**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| N/A | N/A | NO GENERIC CHANGES ARE POSTED AGAINST THIS SPECIFICATION. | Not Applicable | Not Applicable | N/A |

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.5~~A~~ Containment Air Temperature ~~(Atmospheric and Dual)~~

<3.6.C.2> LCO 3.6.5~~A~~ Containment average air temperature shall be ≤ ~~[120]~~ 130 °F.

<3.6.C.2> APPLICABILITY: MODES 1, 2, 3, and 4.

<DOC A.3> ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| <3.6.C.2> A. Containment average air temperature not within limit. | A.1 | Restore containment average air temperature to within limit. | 8 hours |
| <3.6.C.2> B. Required Action and associated Completion Time not met. | B.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | B.2 | Be in MODE 5. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.5~~A~~.1 | Verify containment average air temperature is within limit. | 24 hours |

<3.6.C.3>
<DOC LA.1>
<DOC LA.2>

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5~~X~~ Containment Air Temperature ~~(Atmospheric and Dual)~~

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Containment Spray and Cooling systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

postulated DBAs are analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of the Containment Spray System, Residual Heat Removal System, and Containment Cooling System being rendered inoperable.

may be either a LOCA or a SLB.

The limiting DBA for the maximum peak containment air temperature ~~is an SLB~~. The initial containment average air temperature assumed in the design basis analyses ~~(Ref. 1) is~~, ~~[120]°F. This resulted in a~~ maximum containment air temperature ~~of [384.9]°F. The~~ design temperature ~~is~~ ~~[320]°F~~.

the

and the

are specified in Reference 1.

The temperature limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design temperature for only a few seconds during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA LOCA SLB.

~~The temperature limit is also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System (Ref. 1).~~

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure ~~is a LOCA~~. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of ~~the NRC Policy Statement~~.

10 CFR 50.36

(continued)

BASES (continued)

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Insert B3.6-57-01

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere.

¶ The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. FSAR, Section ~~6.2~~. 14.3
2. 10 CFR 50.49.

**INSERT: B 3.6-55-01**

A representative measurement of containment air temperature requires an arithmetic average of temperatures measured at no fewer than 4 locations. Environmentally and seismically qualified RTDs mounted on the crane wall above the containment fan cooler units inlet are normally used for measuring containment ambient temperature. Portable temperature sensing equipment may also be used.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.5:
# "Containment Air Temperature"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.6.5 - Containment Air Temperature

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.6:

## "Containment Spray System and Containment Fan Cooler System"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.6 CONTAINMENT SYSTEMS

### 3.6.6 Containment Spray System and Containment Fan Cooler System

LCO 3.6.6 Two Containment Spray trains and three Containment Fan Cooler trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One containment spray train inoperable. | A.1 Restore containment spray train to OPERABLE status. | 72 hours<br><br>AND<br><br>10 days from discovery of failure to meet the LCO |
| B. Required Action and associated Completion Time of Condition A not met. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 5. | 6 hours<br><br><br><br>84 hours |

(continued)

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| C. One containment fan cooler train inoperable. | C.1 | Restore containment fan cooler train to OPERABLE status. | 7 days<br><br>AND<br><br>10 days from discovery of failure to meet the LCO |
| D. Two containment fan cooler trains inoperable. | D.1 | Restore one containment fan cooler train to OPERABLE status. | 72 hours |
| E. Required Action and associated Completion Time of Condition C or D not met. | E.1 | Be in MODE 3. | 6 hours |
| | AND | | |
| | E.2 | Be in MODE 5. | 84 hours |
| F. Two containment spray trains inoperable.<br><br>OR<br><br>Any combination of three or more trains inoperable. | F.1 | Enter LCO 3.0.3. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.6.1 | Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position. | 31 days |
| SR 3.6.6.2 | Operate each containment fan cooler unit fan for ≥ 15 minutes. | 92 days |
| SR 3.6.6.3 | Verify each containment fan cooler unit cooling water flow rate is ≥ 1400 gpm. | 92 days |
| SR 3.6.6.4 | Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head. | In accordance with the Inservice Testing Program |
| SR 3.6.6.5 | Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. | 24 months |
| SR 3.6.6.6 | Verify each containment spray pump starts automatically on an actual or simulated actuation signal. | 24 months |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.6.7 | Verify each containment fan cooler unit starts and dampers re-position to the emergency mode automatically on an actual or simulated actuation signal. | 24 months |
| SR 3.6.6.8 | Perform required containment fan cooler system filter testing in accordance with the Ventilation Filter Testing Program (VFTP). | In accordance with the VFTP |
| SR 3.6.6.9 | Verify each spray nozzle is unobstructed. | 10 years |

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.6 Containment Spray System and Containment Fan Cooler System

BASES

BACKGROUND

The Containment Spray System and Containment Fan Cooler System provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Fan Cooler systems are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1).

The Containment Spray System and Containment Fan Cooler System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and the Containment Fan Cooler System provide redundant methods to limit and maintain post accident conditions to less than the containment design values.

Containment Spray System

The Containment Spray System consists of two separate trains. Each train includes a containment spray pump, piping and valves and is independently capable of delivering one-half of the design flow needed to maintain the post-accident containment pressure below 47 psig. The spray water is injected into the containment through spray nozzles connected to four 360 degree ring headers located in the containment dome area. Each train supplies two of the four ring headers. Each train is powered from a separate safeguards power train. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation.

BASES

BACKGROUND

Containment Spray System (Continued)

After the Refueling Water Storage Tank has been exhausted, the containment recirculation pumps or the residual heat removal (RHR) pumps are used to supply the Containment Spray ring headers for the long-term containment cooling and iodine removal during the containment recirculation phase. In this configuration, the RHR heat exchangers provide the necessary cooling of the recirculated containment spray.

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature. Additionally, these systems reduce fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump or recirculation sump water by the residual heat removal heat exchangers. Both trains of the Containment Spray System are needed to provide adequate spray coverage to meet the system design requirements for containment heat removal assuming the Fan Cooler System is not available.

The Spray Additive System injects an NaOH solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated either automatically by a containment High-High pressure signal or manually. An automatic actuation starts the two containment spray pumps, opens the containment spray pump discharge valves, and begins the injection phase. A manual actuation of the Containment Spray System requires the operator to actuate two separate push buttons on the main control board to begin the same sequence. The

BASES

BACKGROUND

Containment Spray System (Continued)

injection phase continues until the RWST water supply is exhausted. After the Refueling Water Storage Tank has been exhausted, the containment recirculation pumps or the residual heat removal (RHR) pumps may be used to supply the Containment Spray ring headers for the long-term containment cooling and iodine removal during the containment recirculation phase. In this configuration, the RHR heat exchangers provide the necessary cooling of the recirculated containment spray. The Containment Spray function in the recirculation mode may be used to maintain an equilibrium temperature between the containment atmosphere and the recirculated sump water. The Containment Spray function in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

Containment Fan Cooler System

The Containment Fan Cooler System consists of five 20% capacity Fan Cooler Units (FCUs) located inside containment. These FCUs are used for both normal and post accident cooling of the containment atmosphere. Each FCU consists of a motor, fan, cooling coils, moisture separators, HEPA filters, carbon filters, dampers, duct distribution system, instrumentation and controls. Service water is supplied to the cooling coils to perform the heat removal function.

During normal plant operation, the moisture separators, HEPA filters and activated carbon filter assembly are isolated from the main air recirculation stream. In this configuration, service water is supplied to all five FCUs and two or more FCUs fans are typically operated to limit the ambient containment air temperature during normal unit operation to less than the limit specified in LCO 3.6.5, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically. Additionally, the actuation signal causes the air flow (air-steam mixture) in each FCU to be split into two parts

BASES

BACKGROUND

Containment Fan Cooler System (continued)

by a bypass flow control damper that fails to a pre-set position for accident operation. A minimum of 8000 cfm is directed through the FCU filtration section (moisture separators, HEPA filters, and carbon filter assembly) with the remainder of the air flow bypassing the filtration section. Both the filtered and unfiltered FCU flow passes through the cooling coils. The temperature of the service water is an important factor in the heat removal capability of the fan units. The accident analysis assumes 1400 gpm of service (cooling) water with a maximum river water inlet temperature of 95° F is supplied to each FCU.

Containment Cooling and Iodine Removal Function

The containment cooling and iodine removal function is provided by either of two systems:

a) the Containment Spray System consisting of two 50% capacity trains; and,

b) The Containment Fan Cooler System consisting of five 20% capacity Fan Cooler Units (FCUs).

Requirements for Containment Spray Trains may be designated by the number of the containment spray pump or the associated safeguards power train. Containment Spray Train 31 is associated with Safeguards Power Train 5A which is supported by DG 33. Containment Spray Train 32 is associated with Safeguards Power Train 6A which is supported by DG 32.

Requirements for the five fan cooler units are designated by grouping the 5 fan cooler units into three trains based on the safeguards power train needed to support Operability. This results in the following designations:

Fan Cooler Train 5A consists of FCU 31 and FCU 33;

Fan Cooler Train 2A/3A consists of FCU 32 and FCU 34; and

Fan Cooler Train 6A consists of FCU 35.

BASES

BACKGROUND

Containment Cooling and Iodine Removal Function (continued)

Design assumptions regarding containment air cooling and iodine removal are met by any of the following configurations:

a) Two containment spray trains; or,

b) Three fan cooler trains (i.e., five fan cooler units); or,

c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units).

This last configuration, one containment spray train and two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure).

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Fan Cooler System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one safeguards power train, which is the worst case single active failure and results in one train of Containment Spray and one train of Fan Coolers being rendered inoperable.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure and temperature may result from either a LOCA or SLB, depending on the cycle specific analysis (Refs. 4 and 6). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion.) The analyses and evaluations assume a unit specific power level of 102% and initial (pre-accident) containment conditions of 130°F

BASES

APPLICABLE SAFETY ANALYSES (continued)

and 2.5 psig and a service water inlet temperature of 95° F. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time includes diesel generator (DG) startup (for loss of offsite power), loading of equipment, containment spray pump startup, and spray line filling.

Containment cooling train performance for post accident conditions is given in References 3, 4 and 6. The result of the analysis is that accident analysis assumptions regarding containment air cooling and iodine removal are met by any of the following configurations:

a) Two containment spray trains; or,

b) Three fan cooler trains (i.e., five fan cooler units); or,

c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units).

BASES

APPLICABLE SAFETY ANALYSES (continued)

This last configuration, one containment spray train and two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure).

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint to achieving full Containment Fan Cooler System air and safety grade cooling water flow. The Containment Cooling System total response time includes signal delay, DG startup (for loss of offsite power), and service water pump startup times (Ref.4).

The Containment Spray System and Containment Fan Cooler System satisfy Criterion 3 of 10 CFR 50.36.

LCO

Accident analysis assumptions regarding containment air cooling and iodine removal are met by any of the following configurations:

a) Two containment spray trains; or,

b) Three fan cooler trains (i.e., five fan cooler units); or,

c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units).

This last configuration, one containment spray train and two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure).

Each Containment Spray System includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal.

Each FCU consists of a motor, fan, cooling coils, moisture separators, HEPA filters, carbon filters, dampers, duct

BASES

LCO (continued)

distribution system, instrumentation and controls necessary to maintain an OPERABLE flow path for the containment atmosphere through both the filtration unit and cooling coils and an OPERABLE flow path for service water through the cooling coils.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and containment cooling trains.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and Containment Fan Cooler System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and fan cooler trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times," for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one of the required containment fan cooler trains inoperable, the inoperable required containment fan cooler train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Fan Cooler System and the low probability of DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action C.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

BASES

ACTIONS (continued)

D.1

With two required containment fan cooler trains inoperable, one of the required containment cooling trains must be restored to OPERABLE status within 72 hours. This allowable out of service time is acceptable because the minimum required containment cooling and iodine removal function is maintained even though this configuration is a substantial degradation from the design capability, and may be a loss of redundancy for this function.

E.1 and E.2

If the Required Action and associated Completion Time of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With two containment spray trains or any combination of three or more containment spray and fan cooler trains inoperable, the unit could be in a condition outside the accident analysis. This Condition ensures that at least one containment spray train and one fan cooler train will be available during an accident. Entering this Condition represents a substantial degradation of the containment heat removal and iodine removal function. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1 (continued)

assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position. Valves in containment with remote position indication may be checked using remote position indication.

SR 3.6.6.2

Operating each containment fan cooler unit for ≥ 15 minutes ensures that all fan cooler units are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 92 day Frequency was developed considering fan coolers are operated during normal plant operation, the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the containment fan cooler units occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying that the service water flow rate to each fan cooler unit is ≥ 1400 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3). The 92 day Frequency was developed considering the known reliability of the Cooling Water System, the redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment High-High pressure signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6.7

This SR requires verification that each containment fan cooler unit starts and damper re-positions to the emergency mode upon receipt of an actual or simulated safety injection signal. The 24 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.7 (continued)

SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 24 month Frequency.

SR 3.6.6.8

This SR verifies that the required Fan Cooler Unit testing is performed in accordance with Specification 5.5.10, Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.6.9

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. 10 CFR 50, Appendix A.
2. 10 CFR 50, Appendix K.
3. FSAR, Sections 6.3 and 6.4.
4. FSAR, Section 14.3.
5. ASME, Boiler and Pressure Vessel Code, Section XI.
6. WCAP - 12269, Containment Margin Improvement Analysis for IP-3 Unit 3, Rev. 1.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.6:

## "Containment Spray System and Containment Fan Cooler System"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.3-5a | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-6 | 145 | 145 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-14 | 132 | 132 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-15 | 139 TSCR 97-175 | 139 TSCR 97-175 | IPN 97-175 | Changes to Bases Pages | |
| 3.3-16 | 154 | 154 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-2 | 172 TSCR 98-043 | 172 TSCR 98-043 | IPN 98-043 | Instrument Channel Surveillance Intervals Extended to 24 Months | Incorporated |
| 4.5-3 | 131 | 131 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-7 | 178 | 178 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-9 | 148 | 148 | No TSCRs | No TSCRs for this Page | N/A |

SEE ITS 3.4.7 and 3.4.8

2) RCS temperature and the source range detectors are monitored hourly;

and

3) no operations are permitted which would reduce the boron concentration of the reactor coolant system.

SEE ITS 3.4.12

8. When the RCS average cold leg temperature ($T_{cold}$) is below 319°F, or when RHR is in service (i.e., not isolated from the RCS), no safety injection pumps shall be energized and aligned to feed the RCS.

9. The requirements of 3.3.A.8 may be relaxed to allow one safety injection pump energized and aligned to feed the RCS under the following circumstances:

a. emergency boration; OR

b. for pump testing, for a period not to exceed 8 hours; OR

c. loss of RHR cooling.

10. The requirements of 3.3.A.8 may be further relaxed when the RCS is < 200°F, such that two safety injection pumps may be energized and aligned to feed the RCS under the following circumstances:

a. the RCS is vented with an opening greater than or equal to the size of one code pressurizer safety valve flange, OR

b.. indicated pressurizer level is at 0% and the plant is vented in accordance with Technical Specification 3.1.A.8.c.1. (Alternate methods and instrumentation may be used to confirm actual RCS elevation.)

LCO 3.6.6

B. Containment Cooling and Iodine Removal Systems A.3

Applicability

1. The ~~reactor shall not be brought above the cold shutdown condition unless~~ Mode 1, 2, 3 and 4 A.4 the following requirements are met:

SEE ITS 3.6.7

a. The spray additive tank contains a minimum of 4000 gallons of solution with a sodium hydroxide concentration ≥35% and ≤38% by weight.

LCO 3.6.6

b. The ~~five fan cooler-charcoal filter units~~ Three containment cooling trains A.3 and the two spray pumps, with their associated valves and piping, are operable.

Action A.1

2. The requirements of 3.3.B.1 may be modified to allow any one A.11 of the following components to be inoperable at one time:

Req. Act C.1

a. ~~Fan cooler unit 32, 34, or 35 or the flow path for fan cooler unit 32, 34, or 35~~ One Fan cooler train may be out of service for a period not to exceed ~~24~~ 7 days hours ~~provided both containment spray pumps are operable~~. (A.11) (L.5)

OR

~~Fan cooler unit 31 or 33, or the flow path for fan cooler unit 31 or 33~~ may be out of service for a period not to exceed 7 days ~~provided both containment spray pumps are operable~~. (A.11)

Req. Act A.1

~~b.~~ One containment spray ~~pump~~ train may be out of service for a period not to exceed ~~24~~ 72 hours, ~~provided the five fan cooler units are operable~~. (A.10) (A.11) (L.1)

~~c.~~ Any valve required for the functioning of the system during and following accident conditions may be inoperable provided it is restored to an operable status within 24 hours and all valves in the system that provide the duplicate function are operable. (A.10)

3. If the Containment Cooling and Iodine Removal are not restored to meet the requirements of 3.3.B.1 within the time period specified in 3.3.B.2, then:

Req. Act B.1, B.2 a. If the reactor is ~~critical~~, it shall be in the hot shutdown condition within ~~four~~ 6 hours and in the cold shutdown condition within ~~the following 24~~ 84 hours. (L.3)

Req. Act E.1, E.2

b. ~~If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.B.1 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.~~ (M.2)

Add Condition D and Req. Act D.1 (L.2)

Add Condition F and Req. Act F.1 (A.5)

Second Completion Time for Req. Act A.1 and C.1 (M.1)

**Bases**

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature, by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.[1] With this mode of startup, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation, and, therefore, the minimum required engineered safeguards and auxiliary cooling systems are required to be operable.

The probability of sustaining both a major accident and a simultaneous failure of a safeguards component to operate as designed is necessarily very small. Thus, operation with the reactor above the cold shutdown condition with minimum safeguards operable for a limited period does not significantly increase the probability of an accident having consequences which are more severe than the Design Basis Accident. A.1

The operable status of the various systems and components is demonstrated by periodic tests defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible, in most cases, to effect repairs and restore the system to full operability within a relatively short time. The inoperability of a single component does not negate the ability of the system to perform its function,[2] but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. Assurance that the redundant component(s) will operate if required to do so exists if the required periodic surveillance testing is current and there are no known reasons to suggest that the redundant component(s) are inoperable. If it develops that (a) the inoperable component is not repaired within the specified allowable time period, or (b) a second component in the same or related system is found to be inoperable, the reactor, if critical, will initially be brought to the hot shutdown condition utilizing normal operating procedures to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. If the reactor was already subcritical, the reactor coolant system temperature and pressure will be maintained within the stated values in order to limit the amount of stored energy in the reactor coolant system. The stated tolerances provide a band for operator control. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the

A.1

cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would damage the fuel elements or result in a release in excess of 10 CFR 100 and 10 CFR 50 dose limits.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and, therefore, in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are based on two considerations:

1) Assuring with high reliability that the safeguard system will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full rated power, the magnitude of the decay heat decreases after initiating hot shutdown. Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 1 hour of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case the reactor is to be put into the cold shutdown condition.

The limits for the Refueling Water Storage Tank and the accumulators insure the required amount of water with the proper boron concentration for injection into the reactor coolant system following a loss-of-coolant accident is available. These limits are based on values used in the accident analysis.(9)(13)

A.1

The minimum indicated RWST level of 35.4 feet (approximately 342,200 gals.), and the low level alarms ("allowable values") of 10.5 feet (approx. 111,100 gals.) and 12.5 feet (approx. 129,700 gals.), include consideration for instrumentation uncertainties, margin, and the unusable volume at the bottom of the tank.(17)(18) These water levels ensure a minimum of approx. 195,800 gals. available for injection, and approx. 66,700 gals. for use during and following the transition from injection to recirculation (to allow continued CS pump operation for sump pH control).(18) The minimum RWST boron concentration ensures that the reactor core will remain subcritical during long term recirculation with all control rods fully withdrawn following a postulated large break LOCA.

The four accumulator isolation valves (894 A,B,C,D) are maintained in the open position when the reactor coolant pressure is above 1000 psig to assure flow passage from the accumulators will be available during the injection phases of a loss-of-coolant accident. Indication is also provided on the monitor light panel, should any of these valves not be in the full open position even with the valve operator deenergized. The 1000 psig limit is derived from the minimum pressure requirements of the accumulators combined with instrument error and an operational band and is based upon avoiding inadvertent injection into the reactor coolant system. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occurring when accumulator core cooling flow is required. Valves 856 B and G are maintained in the closed position to prevent hot leg injection during the injection phase of a loss-of-coolant accident. As an additional assurance of preventing hot leg injection, these valve motor operators are deenergized to prevent spurious opening of these valves during the injection phase of a loss-of-coolant accident. Power will be restored to these valves at an appropriate time in accordance with plant operating procedures after a loss-of-coolant accident in order to establish hot leg recirculation.

Valves 1810, 882, and 744 are maintained in the open position to assure that flow passage from the refueling water storage tank will be available during the injection phase of a loss-of-coolant accident. As additional assurance of flow passage availability, these valve motor operators are de-energized to prevent an extremely unlikely spurious closure. This additional precaution is acceptable, since failure to manually re-establish power to close these valves following the injection phase is tolerable as a single failure.

Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the refueling water storage tank are de-energized in the open position to prevent an extremely unlikely spurious closure which would cause the safety injection pumps to overheat if the reactor coolant system pressure is above the shutoff head of the pumps.

2. Containment Spray System

SR 3.6.6.5 ~~a.~~ System tests shall be performed at least once per 24 months. ~~The tests shall be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed.~~ A.8 Operation of

SR 3.6.6.6 the system is initiated by ~~tripping the normal actuation instrumentation~~ actual or simulated A.9

SR 3.6.6.9 ~~b.~~ The spray nozzles shall be checked for proper functioning at least every ~~five~~ 10 years. L.4

~~c.~~ ~~The tests will be considered satisfactory if visual observations indicate all components have operated satisfactorily.~~ A.8

SEE ITS 3.3.3

3. Containment Hydrogen Monitoring Systems

a. Containment hydrogen monitoring system tests shall be performed at intervals no greater than six months. The tests shall include drawing a sample from the fan cooler units.

b. The above tests will be considered satisfactory if visual observations and control panel indication indicate that all components have operated satisfactorily.

Add SR 3.6.6.1 M.3

Add SR 3.6.6.2 M.4

Add SR 3.6.6.3 M.5



Amendment No. ~~101~~, ~~115~~, ~~125~~, ~~129~~, ~~172~~, TSCR 98-043

Add SR 3.6.6.8



4. Containment Air Filtration System A.6

SEE ITS 5.5.10

a. Visual inspection of the filter installations shall be performed in accordance with ANSI N 510 (1975) every six months for the first two years and at least once per 24 months thereafter. or at any time fire, chemical releases or work done on the filters could alter their integrity.

b. At least once per 24 months, the following conditions shall be demonstrated before the system can be considered operable:

(1) The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at ambient conditions and accident design flow rates.

(2) Using either direct or indirect measurements, the flow rate of the system fans shall be shown to be at least 90% of the accident design flow rate.

SR 3.6.6.7

(3) The charcoal filter isolation valves shall be tested to verify operability. starts automatically on simul or act signal M.6

SEE ITS 5.5.10

c. At least once per 24 months or at any time fire, chemical releases or work done on the filters could alter their integrity or after every 720 hours of charcoal adsorber use since the last test, the following conditions shall be demonstrated before the system can be considered operable:

(1) Impregnated activated charcoal from each of the five units shall have a methyl iodine removal efficiency $\geq$ 85% $\pm$ 20% of the accident design flow rate, 5 to 15 $mg/m^3$ inlet methyl iodine concentration, $\geq$ 95% relative humidity and $\geq$ 250°F. In addition, ignition shall not occur below 300°F.

(2) A halogenated hydrocarbon (freon) test on charcoal adsorbers at $\pm$ 20% of the accident design flow rate and ambient conditions shall show $\geq$ 99% halogenated hydrocarbon removal.

(3) A locally generated DOP* test of the HEPA filters at $\pm$ 20% of the accident design flow rate and ambient conditions shall show $\geq$ 99% DOP removal.

*Dioctylphthalate Particles

SEE CTS MASTER MARKUP

B. Component Tests

1. Pumps

SR 3.6.6.4

~~a.~~ The ~~safety injection pumps, residual heat removal pumps,~~ containment spray pumps and the auxiliary component cooling water pumps shall be started at quarterly intervals. ~~The recirculation pumps shall be started at least once per 24 months.~~ IAW IST Program LA.1

b. Acceptable levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for ~~at least fifteen minutes.~~ A.7

2. Valves

SEE CTS MASTER MARKUP

a. Each spray additive valve shall be cycled by operator action with the pumps shut down at least once per 24 months.

b. The accumulator check valves shall be checked for operability at least once per 24 months.

c. The following check valves shall be checked for gross leakage at least once per 24 months:

| | | | |
|---|---|---|---|
| 857A & G | 857J | 857S & T | 897B |
| 857B | 857K | 857U & W | 897C |
| 857C | 857L | 895A | 897D |
| 857D | 857M | 895B | 838A |
| 857E | 857N | 895C | 838B |
| 857F | 857P | 895D | 838C |
| 857H | 857Q & R | 897A | 838D |

A.1

Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System, and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification 4.1, the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation, and by these means are continuously monitored for satisfactory performance.

The charcoal portion of the containment air recirculation system is a passive safeguard which is isolated from the cooling air flow during normal reactor operation. Hence, the charcoal should have a long useful lifetime. The filter frames that house the charcoal are stainless steel and should also last indefinitely. However, the visual inspection specified in Section A.4(a) of this specification will be performed to verify that this is, in fact, the case. The iodine removal efficiency cannot be measured with the filter cells in place. Therefore, at periodic intervals a representative sample of charcoal is to be removed and tested to verify that the efficiencies for removal of methyl iodide are obtained.[2] The fuel storage building air treatment system is designed to filter the discharge of the fuel storage building atmosphere to the facility vent during normal conditions. As required by Specifications 3.8.A.12 and 3.8.C.6, the fuel storage building emergency ventilation system must be operable whenever irradiated fuel is being moved. However, if the irradiated fuel has had a continuous 45-day decay period, the fuel storage building emergency ventilation system is not technically necessary, even though the system is required to be operable during all fuel handling operations. The emergency ventilation fan is automatically started upon high radiation signal and since the bypass assembly is sealed by manually operated isolation devices, air flow is directed through the emergency ventilation HEPA filters and charcoal adsorbers.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of these adsorbers for all emergency air treatment systems. The charcoal adsorbers are installed to reduce the potential release of radio-iodine to the environment. The in-place test results should indicate a system leak tightness of less than or equal to one percent leakage for the charcoal adsorbers and a HEPA efficiency of greater than or equal to 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of greater than or equal to 90 percent on the fuel handling system samples, and greater than or equal to 85 percent on the containment system samples for expected accident conditions. With the efficiencies of the HEPA filters and charcoal adsorbers as specified, further assurance is provided that the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed.

The basis for the toxic gas monitoring system is given in Technical Specification Section 3.3.

The control room air treatment system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room air treatment system is designed to automatically start upon control room isolation.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.6:

## "Containment Spray System and Containment Fan Cooler System"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.3.B specifies requirements for the Containment Cooling and Iodine Removal Systems which consist of two trains of containment spray and five fan cooler units. ITS LCO 3.6.6 maintains these requirements; however, the LCO name is changed to Containment Spray System and Containment Fan Cooler System because this name is more descriptive and is consistent with the nomenclature used to describe these systems in the FSAR and CTS. Additionally, requirements for the five fan cooler units (i.e., containment fan cooler system) are established by grouping the 5 fan cooler units (FCUs) into three trains based on the safeguards power train needed to support Operability of the fan cooler unit. As a result, FCU 31 and FCU 33 are identified as Containment Fan Cooler train 5A, FCU 32 and FCU 34 are identified as Containment Fan Cooler train 2A/3A, and FCU 35 is identified as Containment Fan Cooler train 6A. This is an administrative change with no adverse impact on safety because there is no change to the existing requirements except as described and justified in this section of the ITS conversion document.

A.4 CTS 3.3.B.1 specifies the Applicability for the Containment Cooling and Iodine Removal Systems as whenever the reactor is above cold shutdown. ITS 3.6.6 maintains this Applicability by requiring that Containment is Operable in Modes 1, 2, 3 and 4 (i.e., whenever the reactor is above cold shutdown). This is an administrative change with no impact on safety because there is no change to the CTS Applicability.

A.5 CTS 3.3.B does not include explicit requirements if the combination of inoperable fan cooler units (i.e., containment fan cooler trains) and/or inoperable containment spray trains result in less than the minimum functional capability assumed in the accident analysis; therefore, CTS 3.0 would require an immediate plant shutdown. This condition would exist if two containment spray trains or three or more trains of spray trains and/or fan cooler trains are inoperable. Under the same conditions, ITS LCO 3.6.6, Condition F and Required Action F.1, requires immediate entry into ITS LCO 3.0.3. This is an administrative change with no adverse impact on safety because there is no change to the existing requirements.

A.6 CTS 4.5.4.a, 4.5.4.b and 4.5.4.c include requirements for the inspection and testing of the containment fan cooler air filtration system. ITS 5.5.10, Ventilation Filter Testing Program, maintains these requirements as part of a Technical Specification program governing the testing of all ventilation filter systems governed by the ITS. ITS SR 3.6.6.8 is added to establish completion of the VFTP as a requirement for the Operability of the containment fan cooler. This is an administrative change with no impact on safety because there is no change to the existing requirements except as identified and justified for ITS 5.5.10.

A.7 CTS 4.5.B.1.a requires each containment spray pump be started periodically and CTS 4.5.B.1.b specifies the test acceptance criteria that each pump starts, reaches the required developed head and "operates for at least 15 minutes." ITS 3.6.6.4 maintains the same requirement; however, the acceptance criterion that the pumps operate for at least 15 minutes is deleted. This change is acceptable because test procedures ensure that stable conditions are established prior to the verification of acceptance criteria and the requirement to operate for 15 minutes does not otherwise contribute to the verification of pump Operability. Therefore, this is an administrative change with no impact on safety.

A.8 CTS 4.5.A.2.a specifies requirements for a functional test of the containment spray system and CTS 4.5.A.2.c establishes the acceptance criteria for this test as "the tests will be considered satisfactory if visual observations indicate all components have operated satisfactorily." ITS SR 3.6.6.5 and SR 3.6.6.6 maintain the requirement for a functional test of the containment spray system; however, the statement that appropriate verification of system performance is limited to visual observations that all components have operated is deleted. This change is acceptable because this type of generic statement is generally not included in the acceptance criteria or either the CTS or ITS. This is an administrative change with no adverse impact on safety because there is no change to the existing requirements.

A.9 CTS 4.5.A.2.a requires a functional test of the containment spray system and specifies that "operation of the system is initiated by tripping the normal actuation instrumentation." ITS SR 3.6.6.5 and SR 3.6.6.6 maintain the requirement for a functional test of the containment spray system; however, the test may be initiated by either an actual or simulated signal. This change is acceptable because use of an actual instead of a simulated or "test" signal will not affect the performance of the test because the equipment being tested cannot discriminate between an actual and simulated signal. This is an administrative change with no impact on safety because the use of an actual or simulated signal does not change the validity of the test as a verification of plant response to the event.

A.10 CTS 3.3.B.2.b provides an allowable out of service time (AOT) one inoperable containment spray pump and CTS 3.3.B.2.c establishes an allowable out of service time of 24 hours for any valve required for the functioning of the system (i.e., core spray system). Under ITS LCO 3.6.6, requirements are established on the basis of containment spray trains and a train is considered inoperable if either a pump or a valve associated with that pump is inoperable. This is an administrative change with no adverse impact on safety because there are no changes to the existing requirements except as identified and justified elsewhere in this document.

A.11 CTS 3.3.B.2 specifies that "any one" of the five fan cooler units or containment spray pumps to be inoperable at any one time and CTS 3.3.B.2.a, CTS 3.3.B.2.b, and CTS 3.3.B.2.c do not permit any allowable out of service time (AOT) if redundant trains of containment spray or fan cooler units are inoperable.

ITS LCO 3.6.6 allows an allowable out of service time (AOT) even if redundant trains of containment spray or fan cooler units are inoperable as long as the combination of inoperable fan cooler units (i.e., containment fan cooler trains) and/or inoperable containment spray trains do not result in less than the minimum functional capability assumed in the accident analysis.

This is an administrative change with no adverse impact on safety because there are no changes to the existing requirements except as identified and justified in ITS 3.6.6, DOCs L.1, L.2 and L.5.

MORE RESTRICTIVE

M.1 CTS 3.3.B.2 establishes allowable out of service times for the Containment Cooling and Iodine Removal System; however, there is no limit on the maximum amount of time that any combination of containment spray trains and containment fan cooler trains may be out of service. ITS LCO 3.6.6, Required Actions A.1 and C.1, have new supplementary Completion Times that establish a limit on the maximum consecutive time that the plant may be without the full complement of containment cooling and iodine removal capability. This supplementary Completion Time is needed to place a reasonable limit on the amount of time that operation may continue with degraded containment cooling and iodine removal capability consistent with the intent of the Allowable Out of Service Times (AOTs) for a single train or other LCO 3.6.6 Condition. This change is acceptable because it does not introduce any operation which is un-analyzed while placing a reasonable limit on the amount of time that Operation may continue with degraded containment cooling and iodine removal capability. Therefore, this change has no adverse impact on safety.

M.2 CTS 3.3.B.3 establishes the Actions required if either containment spray and/or containment fan cooler trains are not restored to meet CTS requirements within specified completion times. CTS 3.3.B.3.a specifies that, if the reactor is critical when requirements are not met, then the reactor must be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within the following 24 hours. However, if the reactor is subcritical when requirements are not met, CTS 3.3.B.3.b requires only that reactor coolant system temperature and pressure not be increased more than 25°F and 100 psi, respectively, over existing values with the requirement to proceed to cold shutdown (Mode 5) deferred by 48 hours.

Under the same conditions, ITS 3.6.6, Required Actions B.1 and B.2 and/or Required Actions E.1 and E.2, require that the reactor be in Mode 3 in 6 hours (See ITS 3.6.6, DOC L.3) and in Mode 5 in 84 hours (Required Actions B.2,and E.2, See ITS 3.6.6, DOC L.3), regardless of the status of the unit when the Condition is identified. The allowance provided in CTS 3.3.B.3.b is deleted.

This change is needed to eliminate the ambiguity created by CTS 3.3.B.3.b when performing a reactor shutdown and cooldown required by CTS 3.3.B.3.a and to ensure that the plant is placed outside the LCO Applicability promptly when the LCO requirements are not met. This change is acceptable because placing the plant outside the LCO Applicability when LCO requirements are not met is conservative and there is no change in the CTS 3.3.B.3 requirement. This change has no significant adverse impact on safety.

M.3 ITS SR 3.6.6.1 is added to require verification every 31 days that each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position. There is no equivalent requirement in the CTS. This change is needed because it provides assurance that the proper flow paths exist for containment spray system operation, and that any valves outside containment that are capable of potentially being mispositioned are in the correct position. This change has no adverse impact on safety.

M.4 ITS SR 3.6.6.2 is added to require operation of each required containment fan cooler train for ≥ 15 minutes every 92 days. There is no equivalent requirement in the CTS. This change is needed because it provides assurance that all trains are operable and all controls are functioning properly and it ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. This change has no adverse impact on safety.

M.5 ITS SR 3.6.6.3 is added to require verification every 24 months that cooling water flow to each fan cooler unit is ≥ 1400 gpm. There is no equivalent requirement in the CTS. This change is needed because it provides assurance that the design flow rate assumed in the safety analysis will be achieved during an accident. This change has no adverse impact on safety.

M.6 CTS 4.5.A.4.b.3 requires that charcoal filter (i.e., fan cooler unit) isolation valves shall be tested to verify operability every 24 months. ITS SR 3.6.6.7 expands this surveillance to require verification that each fan cooler unit actuates and dampers re-position on receipt of an actual or simulated safety injection signal. This change is needed because it provides assurance that all fan cooler units will start and dampers re-position when required to mitigate a design basis accident. This change has no adverse impact on safety.

## LESS RESTRICTIVE

L.1 CTS 3.3.B.2.b provides an allowable out of service time (AOT) of 24 hours for one inoperable containment spray pump; and, CTS 3.3.B.2.c establishes an allowable out of service time of 24 hours for any valve required for the functioning of the system (i.e., core spray system) provided all valves in the system that provide the duplicate function are operable.

Under the same conditions, ITS LCO 3.6.6 establishes an AOT of 72 hours for one inoperable containment spray train (i.e., one pump and/or any associated valve). This change is needed because 24 hours is more restrictive than the AOT for the associated safeguards power train. This change is acceptable because the 72-hour completion time takes into account the redundant heat removal capability (i.e., the redundant containment spray train and the fan cooler trains), it provides a reasonable time for repairs, and it is consistent with the low probability of a design basis accident occurring during this period. Therefore, this change has no significant adverse impact on safety.

L.2 CTS 3.3.B.2 provides allowable out of service times (AOTs) for fan cooler units (FCUs); however, there is no AOT when more than one FCU is inoperable. Therefore, if more than one FCU is inoperable, then a plant shutdown must be initiated.

ITS LCO 3.6.6, Condition D, provides an AOT of 72 hours if two containment fan cooler trains are inoperable (except if ITS LCO 3.6.6, Condition F, is entered when any combination of three or more trains are inoperable or when two containment spray trains are inoperable). This change is acceptable because Condition D represents a loss of redundancy but the minimum required containment cooling and iodine removal function is maintained. Additionally, Condition F ensures there is no loss of containment cooling and iodine removal function. This is true because any of the following combinations of equipment will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value: a) Two containment spray trains; or, b) Three fan cooler trains (i.e., all five fan cooler units); or, c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units). This last configuration, one containment spray train and any two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure). Additionally, the 72-hour AOT for loss of redundancy for the containment cooling and iodine removal function is bounded by the AOT for an inoperable diesel generator and takes into account the redundant heat removal capabilities, the iodine removal function of the Containment Spray System, and the low probability of DBA occurring during this period. Therefore, this change has no significant adverse impact on safety.

L.3 CTS 3.3.B.3 establishes the Actions required if either containment spray and/or containment fan cooler trains are not restored to meet CTS requirements within specified completion times. CTS 3.3.B.3.a specifies that, if the reactor is critical when requirements are not met, then the reactor must be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within the following 24 hours (See ITS 3.6.6, DOC M.2). Under the same conditions, ITS 3.6.6, Required Actions B.1 and B.2 and/or E.1 and E.2, require that the reactor be in Mode 3 in 6 hours and in Mode 5 in 84 hours.

This change is needed and is acceptable because placing the reactor in Mode 3 in 6 hours and in Mode 5 in 84 hours is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach Mode 5 when one containment spray train or fan cooler unit is inoperable provides additional time for attempting restoration when there is minimal loss of capacity. This Completion Time is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in Mode 3. Therefore, this change has no significant adverse impact on safety.

L.4 CTS 4.5.2.b requires that the spray nozzles be checked for proper functioning at least every five years. ITS SR 3.6.6.requires verification every ten years that each spray nozzle is unobstructed. This change, extending the SR Frequency from 5 years to 10 years, is acceptable because the spray nozzles are passive devices and industry experience indicates this interval is sufficient to detect obstruction of the spray nozzles. During the last two performances of this SR, there was no evidence of obstruction or improper functioning of the spray nozzles. Therefore, this change has no adverse impact on safety.

L.5 CTS 3.3.B.2.a provides allowable out of service times for fan cooler units (FCUs) as follows:

a. states that FCU 32, 34, or 35, or the flow path for FCU 32, 34, or 35 may be out of service for a period not to exceed 24 hours, provided both containment spray pumps are operable; or,

b. FCU 31 or 33, or the flow path for FCU 31 or 33, may be out of service for 7 days, provided both containment spray pumps are operable.

This set of Actions will allow only one of the five FCUs to be inoperable at one time and then only if no containment spray train in inoperable.

ITS LCO 3.6.6 establishes requirements for three Fan Cooler System trains where FCU 31 and FCU 33 are identified as Containment Fan Cooler train 5A, FCU 32 and FCU 34 are identified as Containment Fan Cooler train 2A/3A, and FCU 35 is identified as Containment Fan Cooler train 6A (See ITS 3.6.6, DOC A.3).

ITS LCO 3.6.6, Required Action C.1, allows any one train (i.e., up to 2 FCUs) to be inoperable for 7 days and ITS LCO 3.6.6, Required Action D.1, allows any two trains (i.e., up to 4 FCUs) to be inoperable for 72 hours. This change is acceptable because Conditions C and D represent a loss of redundancy but the minimum required containment cooling and iodine removal function is maintained and Condition F (Enter LCO 3.0.3 if any combination of three or more trains of containment spray or FCUs are inoperable). Therefore, Required Actions C.1, D.1 and F.1 ensure there is no loss of containment cooling and iodine removal function because any of the following combinations of equipment will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value: a) Two containment spray trains; or, b) Three fan cooler trains (i.e., all five fan cooler units); or, c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units). This last configuration, one containment spray train and any two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure). Additionally, the 72-hour AOT for loss of redundancy for the containment cooling and iodine removal function is bounded by the AOT for an inoperable diesel generator and takes into account the redundant heat removal capabilities, the iodine removal function of the Containment Spray System, and the low probability of DBA occurring during this period. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

LA.1 CTS 4.5.B requires that the containment spray pumps be started every quarter. ITS 3.6.6.4 maintains the same requirement except that the SR Frequency is established by the Inservice Testing Program.

This change is acceptable because the IST Program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components and is required by ITS 5.5.7. ITS 5.5.7, Inservice Testing Program (IST), requires establishing and maintaining a program for inservice testing of ASME Code Class 1, 2, and 3 components at frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code. Additionally, 10 CFR 50.55a(f) already provides the regulatory requirements for this IST Program, and specifies that ASME Code Class 1, 2, and 3 pumps and valves are covered by an IST Program. Therefore, maintaining the requirement that containment spray trains must be Operable in ITS 3.6.6 and maintaining the requirement for periodic testing of pumps and valves in the IST Program required by ITS 5.5.7 provides a high degree of assurance that check valves will be tested and maintained to ensure containment spray Operability. Additionally, ITS 5.5.7, Inservice Testing Program (IST), requirements and 10 CFR 50.55a(f) ensure adequate change control and regulatory oversight for any changes to the existing requirements. Therefore, the testing Frequency for containment spray can be maintained in the IST program with no significant adverse impact on safety.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.6:

# "Containment Spray System and Containment Fan Cooler System"

## PART 4:

## No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.6.6 - Containment Spray System and Containment Fan Cooler System

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   CTS 3.3.B.2.b provides an allowable out of service time (AOT) of 24 hours for one inoperable containment spray pump and CTS 3.3.B.2.c establishes an allowable out of service time of 24 hours for any valve required for the functioning of the system (i.e., core spray system) provided all valves in the system that provide the duplicate function are operable. Under the same conditions, ITS LCO 3.6.6 establishes an AOT of 72 hours for one inoperable containment spray train (i.e., one pump and/or any associated valve).

   This change will not result in a significant increase in the probability of an accident previously evaluated because containment spray system status is not the initiator of any analyzed event. This change will not result in a significant increase in the consequences of an accident previously evaluated because the 72-hour completion time takes into account the redundant heat removal capability (i.e., the redundant containment spray train and the fan cooler trains), it provides a reasonable time for repairs, and it is consistent with the low probability of a design basis accident occurring during this period.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because the containment cooling and air filtration trains are for accident mitigation, and the remaining operable spray and cooling and air filtration trains are adequate to perform the iodine removal and containment cooling functions.

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LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

CTS 3.3.B.2 provides allowable out of service times (AOTs) for fan cooler units (FCUs); however, there is no AOT when more than one FCU is inoperable. Therefore, if more than one FCU is inoperable, then a plant shutdown must be initiated. ITS LCO 3.6.6, Condition D, provides an AOT of 72 hours if two containment fan cooler trains are inoperable (except if ITS LCO 3.6.6, Condition F, is entered when any combination of three or more trains are inoperable or when two containment spray trains are inoperable).

This change will not result in a significant increase in the probability of an accident previously evaluated because FCU status is not the initiator of any analyzed event.

This change will not result in a significant increase in the consequences of an accident previously evaluated because Condition D represents a loss of redundancy but the minimum required containment cooling and iodine removal function is maintained and Condition F ensures there is no loss of containment cooling and iodine removal function. This is because any of the following combinations of equipment will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value: a) Two containment spray trains; or, b) Three fan cooler trains (i.e., all five fan cooler units); or, c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units). This last configuration, one containment spray train and any two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure). Additionally, the 72-hour AOT for loss of redundancy for the containment cooling and iodine removal function is bounded by the AOT for an inoperable diesel generator and takes into account the redundant heat removal capabilities, the iodine removal function of the Containment Spray System, and the low probability of DBA occurring during this period.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the Condition represents a loss of redundancy but the minimum required containment cooling and iodine removal function is maintained and Condition F ensures there is no loss of containment cooling and iodine removal function.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.6.6 - Containment Spray System and Containment Fan Cooler System

LESS RESTRICTIVE
("L.3" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   CTS 3.3.B.3 establishes the Actions required if either containment spray and/or containment fan cooler trains are not restored to meet CTS requirements within specified completion times. CTS 3.3.B.3.a specifies that, if the reactor is critical when requirements are not met, then the reactor must be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within the following 24 hours (See ITS 3.6.6, DOC M.2). Under the same conditions, ITS 3.6.6, Required Actions B.1 and B.2 and/or Required Actions E.1 and E.2, require that the reactor be in Mode 3 in 6 hours and in Mode 5 in 84 hours (Required Action B.2 and (Required Action E.2).

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because placing the reactor in Mode 3 in 6 hours and in Mode 5 in 84 hours is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. Additionally, the extended interval to reach Mode 5 when one containment spray train or fan cooler unit is inoperable is inoperable provides additional time for attempting restoration of the containment spray train or fan cooler unit. This Completion Time is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in Mode 3.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because placing the reactor in Mode 3 in 6 hours and in Mode 5 in 84 hours is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. Additionally, the extended interval to reach Mode 5 when one containment spray train or fan cooler unit is inoperable is inoperable provides additional time for attempting restoration of the containment spray train or fan cooler unit. This Completion Time is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in Mode 3.

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LESS RESTRICTIVE
("L.4" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   CTS 4.5.2.b requires that the spray nozzles be checked for proper functioning at least every five years. ITS SR 3.6.6.requires verification every ten years that each spray nozzle is unobstructed.

This change will not result in a significant increase in the probability of an accident previously evaluated because the status of the spray nozzles is not the initiator of any analyzed event. This change will not result in a significant increase in the consequences of an accident previously evaluated because the spray nozzles are passive devices and industry experience indicates this interval is sufficient to detect obstruction of the spray nozzles. This surveillance has been performed twice at IP3 with no evidence of obstruction or improper functioning of the spray nozzles.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the spray nozzles are passive devices and industry experience indicates this interval is sufficient to detect obstruction of the spray nozzles.

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LESS RESTRICTIVE
("L.5" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   CTS 3.3.B.2.a provides allowable out of service times for fan cooler units (FCUs) that allow only one of the five FCUs to be inoperable at one time and then only if no containment spray train in inoperable.

   ITS LCO 3.6.6 establishes requirements for three Fan Cooler System trains where FCU 31 and FCU 33 are identified as Containment Fan Cooler train 5A, FCU 32 and FCU 34 are identified as Containment Fan Cooler train 2A/3A, and FCU 35 is identified as Containment Fan Cooler train 6A (See ITS 3.6.6, DOC A.3).

   ITS LCO 3.6.6, Required Action C.1, allows any one train (i.e., up to 2 FCUs) to be inoperable for 7 days and ITS LCO 3.6.6, Required Action D.1, allows any two trains (i.e., up to 4 FCUs) to be inoperable for 72 hours.

   This change will not result in a significant increase in the probability of an accident previously evaluated because fan cooler unit status is not the initiator of any analyzed event; therefore, the proposed change to the actions when this limit is not met is not the initiator of any analyzed event. This change will not result in a significant increase in the consequences of an accident previously evaluated because this condition represents a loss of redundancy but the minimum required containment cooling and iodine removal function is maintained. Therefore, Required Actions C.1, D.1 and F.1 ensure there is no loss of containment cooling and iodine removal function because any of the following combinations of equipment will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value: a) Two containment spray trains; or, b) Three fan cooler trains (i.e., all five fan cooler units); or, c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units). This last configuration, one containment spray train and any two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure). Additionally, the

72-hour AOT for loss of redundancy for the containment cooling and iodine removal function is bounded by the AOT for an inoperable diesel generator and takes into account the redundant heat removal capabilities, the iodine removal function of the Containment Spray System, and the low probability of DBA occurring during this period.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because this condition represents a loss of redundancy but the minimum required containment cooling and iodine removal function is maintained. Therefore, Required Actions C.1, D.1 and F.1 ensure there is no loss of containment cooling and iodine removal function.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.6:

## "Containment Spray System and Containment Fan Cooler System"

## PART 5:

## NUREG-1431
## Annotated to show differences between NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.6.6

This ITS Specification is based on NUREG-1431 Specification No. **3.6.6A**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| BWOG-001 | | REVISE CONTAINMENT COOLING SYSTEM FLOW RATE SR TO TEST HEAT REMOVAL | Rejected by TSTF | Not Incorporated | N/A |

Insert: 3.6-23-01

3.6 CONTAINMENT SYSTEMS

<CTS>

3.6.6~~A~~ Containment Spray and Cooling Systems ~~(Atmospheric and Dual)~~
~~(Credit taken for iodine removal by the Containment Spray System)~~

<3.3.B.1.b> <Doc A.3> LCO 3.6.6~~A~~ Two containment spray trains and ~~two~~ Three containment ~~cooling~~ fan cooler trains shall be OPERABLE.

<3.3.B> <Doc A.4> APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <3.3.B.2.b> <Doc L.1> <Doc A.10> <Doc M.1> | A. One containment spray train inoperable. | A.1 Restore containment spray train to OPERABLE status. | 72 hours<br>AND<br>10 days from discovery of failure to meet the LCO |
| <3.3.B.3.a> <Doc L.3> | B. Required Action and associated Completion Time of Condition A not met. | B.1 Be in MODE 3.<br>AND<br>B.2 Be in MODE 5. | 6 hours<br><br>84 hours |
| <3.3.B.2.a> <Doc L.5> <Doc M.1> | C. One ~~[required]~~ containment ~~cooling~~ fan cooler train inoperable. | C.1 Restore ~~[required]~~ containment ~~cooling~~ fan cooler train to OPERABLE status. | ~~7 days~~ 7 days<br>AND<br>10 days from discovery of failure to meet the LCO |

(continued)

**INSERT: 3.6-23-01**

Containment Spray System and Containment Fan Cooler System

ACTIONS (continued)

| <CTS> | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <3.3.B.2.a> <DOC L.2> | D. Two ~~[required]~~ containment cooling trains inoperable. fan cooler | D.1 Restore one ~~[required]~~ containment cooling train to OPERABLE status. | 72 hours |
| <3.3.B.3.a> <DOC L.3> <DOC M.2> | E. Required Action and associated Completion Time of Condition C or D not met. | E.1 Be in MODE 3.<br>AND<br>E.2 Be in MODE 5. | 6 hours<br><br>~~36~~ 84 hours |
| <DOC A.5> | F. Two containment spray trains inoperable.<br>OR<br>Any combination of three or more trains inoperable. | F.1 Enter LCO 3.0.3. | Immediately |

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | | FREQUENCY |
|---|---|---|---|
| <DOC M.3> | SR 3.6.6~~E~~.1 | Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position. | 31 days |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| | SURVEILLANCE | | FREQUENCY | |
|---|---|---|---|---|
| <DOC M.4> | SR 3.6.6.2 | Operate each ~~[required]~~ containment cooling train fan unit for ≥ 15 minutes. unit fan cooler | 31 days 92 | CLB.1 |
| <DOC M.5> | SR 3.6.6.3 | Verify each ~~[required]~~ containment cooling train cooling water flow rate is ≥ 700 gpm. 1400 fan cooler unit | 31 days 92 | CLB.1 |
| <4.5.B.1.a> <4.5.B.1.b> <DOC A.7> | SR 3.6.6.4 | Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head. | In accordance with the Inservice Testing Program | |
| <4.5.A.2.a> <4.5.A.2.c> <DOC A.8> | SR 3.6.6.5 | Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. | 18 months 24 | |
| <4.5.A.2.a> <4.5.A.2.c> <DOC A.9> <DOC A.8> | SR 3.6.6.6 | Verify each containment spray pump starts automatically on an actual or simulated actuation signal. | 18 months 24 | |
| <4.5.A.4.b.3> <DOC M.6> | SR 3.6.6.7 | Verify each ~~[required]~~ containment cooling train starts automatically on an actual or simulated actuation signal. fan cooler unit and dampers re-position to the emergency mode | 18 months 24 | |

Insert: 3.6-25-01

(continued)

NUREG-1431 Markup Inserts

ITS SECTION 3.6.6 - Containment Spray System and Containment Fan Cooler System

<u>INSERT: 3.6-25-01:</u>

| | | |
|---|---|---|
| SR 3.6.6.8<br><CTS><br><4.5.4><br><DOC A.6> | Perform required containment fan cooler system filter testing in accordance with the Ventilation Filter Testing Program (VFTP). | In accordance with the VFTP |

SURVEILLANCE REQUIREMENTS (continued)

<4.5.A.2.b>
<DOC L.4>

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.6.6~~A.8~~ 9 Verify each spray nozzle is unobstructed. | ~~At first refueling~~ ~~AND~~ 10 years |

Insert: B 3.6-64-01

## B 3.6 CONTAINMENT SYSTEMS

## B 3.6.6A Containment Spray and Cooling Systems ~~(Atmospheric and Dual)~~ ~~(Credit taken for iodine removal by the Containment Spray System)~~

## BASES

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**BACKGROUND**

The Containment Spray and Containment Cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling systems are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1), or other documents that were appropriate at the time of licensing (identified on a unit specific basis).

Fan Cooler

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and the Containment Cooling System provide redundant methods to limit and maintain post accident conditions to less than the containment design values.

Fan Cooler

### Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump

Insert: B 3.6-64-02

Insert: B 3.6-64-03

Safeguards power train

(continued)

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INSERT: 3.6-64-01

Containment Spray System and Containment Fan Cooler System

INSERT: 3.6-64-02

Each train includes a containment spray pump, piping and valves and is independently capable of delivering one-half of the design flow needed to maintain the post-accident containment pressure below 47 psig. The spray water is injected into the containment through spray nozzles connected to four 360 degree ring headers located in the containment dome area. Each train supplies two of the four ring headers.

INSERT: 3.6-64-03

After the Refueling Water Storage Tank has been exhausted, the containment recirculation pumps or the Residual Heat Removal (RHR) pumps are used to supply the Containment Spray ring headers for the long-term containment cooling and iodine removal during the containment recirculation phase. In this configuration, the RHR heat exchangers provide the necessary cooling of the recirculated containment spray.

BASES

BACKGROUND

Containment Spray System (continued)

~~suction is transferred from the RWST to the Containment sump(s).~~

Additionally, these systems

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature. and to reduce fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the residual heat removal ~~coolers~~. ~~Each train~~ of the Containment Spray System provide~~s~~ adequate spray coverage to meet the system design requirements for containment heat removal.

are needed to

heat exchangers

Both trains

or recirculation sump

assuming the Fan Cooler System is not available

The Spray Additive System injects an NaOH solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated either automatically by a containment ~~High-3~~ pressure signal or manually. An automatic actuation ~~opens the containment spray pump discharge valves~~, starts the two containment spray pumps, and begins the injection phase. A manual actuation of the Containment Spray System requires the operator to actuate two separate ~~switches~~ on the main control board to begin the same sequence. The injection phase continues until ~~an RWST level Low-Low alarm is received. The Low-Low level alarm for the RWST actuates valves to align the Containment Spray System pump suction with the containment sump and/or signals the operator to manually align the system to the recirculation mode.~~ The Containment Spray ~~System~~ in the recirculation mode maintain~~s~~ an equilibrium temperature between the containment atmosphere and the recirculated sump water. ~~Operation of the~~ The Containment Spray ~~System~~ in the recirculation mode is

High-High

push buttons

Insert: B 3.6-65-01

may be used to

function

(continued)

**INSERT: 3.6-65-01**

the RWST water supply is exhausted. After the Refueling Water Storage Tank has been exhausted, the containment recirculation pumps or the Residual Heat Removal (RHR) pumps may be used to supply the Containment Spray ring headers for the long-term containment cooling and iodine removal during the containment recirculation phase. In this configuration, the RHR heat exchangers provide the necessary cooling of the recirculated containment spray.

BASES

BACKGROUND

Containment Spray System (continued)

controlled by the operator in accordance with the emergency operating procedures.

Fan Cooler

Containment Cooling System

Insert: B3.6-66-01

~~Two trains of containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement, are provided. Each train of two fan units is supplied with cooling water from a separate train of essential service water (ESW). Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, and instrument tunnel, and outside the secondary shield in the lower areas of containment.~~

~~During normal operation, all four fan units are operating. The fans are normally operated at high speed with ESW supplied to the cooling coils. The Containment Cooling System, operating in conjunction with the Containment Ventilation and Air Conditioning systems, is designed~~ to limit the ambient containment air temperature during normal unit operation to less than the limit specified in LCO 3.6.5~~A~~, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

Insert B3.6-66-04

In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically in slow speed ~~if not already running~~. ~~If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere.~~ The temperature of the ESW is an important factor in the heat removal capability of the fan units.

Insert: B3.6-66-02

Insert: B3.6-66-03

Service Water

APPLICABLE SAFETY ANALYSES

Insert: B3.6-66-05

The ~~Containment Spray System and Containment Cooling System~~ limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed

(continued)

**INSERT: 3.6-66-01**

The Containment Fan Cooler System consists of five 20% capacity Fan Cooler Units (FCUs) located inside containment. These FCUs are used for both normal and post accident cooling of the containment atmosphere. Each FCU consists of a motor, fan, cooling coils, moisture separators, HEPA filters, carbon filters, dampers, duct distribution system, instrumentation and controls. Service water is supplied to the cooling coils to perform the heat removal function.

During normal plant operation, the moisture separators, HEPA filters and activated carbon filter assembly are isolated from the main air recirculation stream. In this configuration, service water is supplied to all five FCUs and two or more FCUs fans are typically operated

**INSERT: 3.6-66-02**

Additionally, the actuation signal causes the air flow (air-steam mixture) in each FCU to be split into two parts by a bypass flow control damper that fails to a pre-set position for accident operation. A minimum of 8000 cfm is directed through the FCU filtration section (moisture separators, HEPA filters, and carbon filter assembly) with the remainder of the air flow bypassing the filtration section. Both the filtered and unfiltered FCU flow passes through the cooling coils.

**INSERT: 3.6-66-03**

The accident analysis assumes 1400 gpm of service (cooling) water with a maximum river water inlet temperature of 95° F is supplied to each FCU.

<u>INSERT: 3.6-66-04</u>

<u>Containment Cooling and Iodine Removal Function</u>

The containment cooling and iodine removal function is provided by either of two systems;

a) the Containment Spray System consisting of two 50% capacity trains; and,
b) The Containment Fan Cooler System consisting of five 20% capacity Fan Cooler Units (FCUs).

Requirements for Containment Spray Trains may be designated by the number of the containment spray pump or the associated safeguards power train. Containment Spray Train 31 is associated with Safeguards Power Train 5A which is supported by DG 33. Containment Spray Train 32 is associated with Safeguards Power Train 6A which is supported by DG 32.

Requirements for the five fan cooler units are designated by grouping the 5 fan cooler units into three trains based on the safeguards power train needed to support Operability. This results in the following designations:

Fan Cooler Train 5A consists of FCU 31 and FCU 33;
Fan Cooler Train 2A/3A consists of FCU 32 and FCU 34; and
Fan Cooler Train 6A consists of FCU 35.

Design assumptions regarding containment air cooling and iodine removal are met by any of the following configurations:

a) Two containment spray trains; or,
b) Three fan cooler trains (i.e., five fan cooler units); or,
c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units).

This last configuration, one containment spray train and two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure).

<u>INSERT: 3.6-66-05</u>

Containment Spray System and Containment Fan Cooler System

APPLICABLE SAFETY ANALYSES (continued)

to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train of ~~the~~ Containment Spray ~~System~~ and Containment Cooling System being rendered inoperable.

Safeguards power train

one train of Fan Coolers

The analysis and evaluation show that under the worst case scenario, the highest peak containment ~~pressure is [44.1] psig (experienced during a LOCA). The analysis shows that the peak containment temperature is [384.5]°F (experienced during an SLB).~~ Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4~~A~~, "Containment Pressure," and LCO 3.6.5~~A~~ for a detailed discussion.) The analyses and evaluations assume a unit specific power level of [100]%, ~~one containment spray train and one containment cooling train operating,~~ and initial (pre-accident) containment conditions of [120]°F and [1.5] psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

Insert: B 3.6-67-01

102

2.5

130

Insert: B 3.6-67-02

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a ~~[2.0] psig~~ containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4~~A~~.

rapid reduction of

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment ~~High-3~~ pressure setpoint to achieving full flow through the containment spray nozzles.

High High

(continued)

**INSERT: 3.6-67-01**

pressure and temperature may result from either a LOCA or SLB, depending on the cycle specific analysis (Refs. 4 and 6)

**INSERT: 3.6-67-02**

and a service water inlet temperature of 95° F.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Containment Spray System total response time of [60] seconds includes diesel generator (DG) startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling ~~(Ref. 3)~~.

References 3, 4 and 6

Containment cooling train performance for post accident conditions is given in Reference 4. The result of the analysis is that ~~each train can provide 100% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 5.~~

Insert: B 3.6-68-01

Fan Cooler

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-3 pressure setpoint to achieving full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time ~~of [60] seconds~~ includes signal delay, DG startup (for loss of offsite power), and service water pump startup times (Ref. 6). 4

High High

Insert: B 3.6-68-02

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of ~~the NRC Policy Statement~~.

10 CFR 50.36

LCO

~~During a DBA, a minimum of one containment cooling train and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Ref. 7). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling trains must be OPERABLE. Therefore, in the event of an accident, at least one train in each system operates, assuming the worst case single active failure occurs.~~

Insert: B 3.6-68-03

Each Containment Spray System typically ~~includes~~ a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal. ~~and automatically transferring suction to the containment sump.~~

(continued)

INSERT: 3.6-68-01

Accident analysis assumptions regarding containment air cooling and iodine removal are met by any of the following configurations:

a) Two containment spray trains; or,
b) Three fan cooler trains (i.e., five fan cooler units); or,
c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units).

This last configuration, one containment spray train and two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure).

INSERT: 3.6-68-02

Containment Spray System and Containment Fan Cooler System

INSERT: 3.6-68-03

Accident analysis assumptions regarding containment air cooling and iodine removal are met by any of the following configurations:

a) Two containment spray trains; or,
b) Three fan cooler trains (i.e., five fan cooler units); or,
c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units).

This last configuration, one containment spray train and two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure).

Insert: B 3.6-69-01

BASES

LCO (continued)

~~Each Containment Cooling System typically includes demisters, cooling coils, dampers, fans, instruments, and controls to ensure an OPERABLE flow path.~~

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and containment cooling trains.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the ~~Containment Spray System and the Containment Cooling System~~ are not required to be OPERABLE in MODES 5 and 6.

Insert: B 3.6-69-02

ACTIONS

A.1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and ~~cooling~~ Fan Cooler trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times," for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the

(continued)

**INSERT: 3.6-69-01**

Each FCU consists of a motor, fan, cooling coils, moisture separators, HEPA filters, carbon filters, dampers, duct distribution system, instrumentation and controls necessary to maintain an OPERABLE flow path for the containment atmosphere through both the filtration unit and cooling coils and an OPERABLE flow path for service water through the cooling coils.

**INSERT: 3.6-69-02**

Containment Spray System and Containment Fan Cooler System

BASES

ACTIONS

B.1 and B.2 (continued)

plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

fan cooler

With one of the required containment cooling trains inoperable, the inoperable required containment cooling train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the ~~Containment Spray System and Containment Cooling System~~ and the low probability of DBA occurring during this period.

Insert: B3.6-70-01

The 10 day portion of the Completion Time for Required Action C.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

D.1

fan cooler

With two required containment cooling trains inoperable, one of the required containment cooling trains must be restored to OPERABLE status within 72 hours. ~~The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal~~

Insert: B3.6-70-02

(continued)

INSERT: 3.6-70-01

Containment Spray System and Containment Fan Cooler System

INSERT: 3.6-70-02

This allowable out of service time is acceptable because the minimum required containment cooling and iodine removal function is maintained even though this configuration is a substantial degradation from the design capability, and may be a loss of redundancy for this function.

BASES

ACTIONS

D.1 (continued)

~~needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System, the iodine removal function of the Containment Spray System, and the low probability of DBA occurring during this period.~~

E.1 and E.2

If the Required Action and associated Completion Time of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. (84)

F.1

Insert: B 3.6-71-01

fan cooler

With two containment spray trains or any combination of three or more containment spray and ~~cooling~~ trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

could be

SURVEILLANCE REQUIREMENTS

SR 3.6.6~~A~~.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment ~~(only check valves are inside containment)~~ and capable of potentially being mispositioned are in the correct position.

Insert: B 3.6-71-02

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.6.6 - Containment Spray System and Containment Fan Cooler System

INSERT: 3.6-71-01

This Condition ensures that at least one containment spray train and one fan cooler train will be available during an accident. Entering this Condition represents a substantial degradation of the containment heat removal and iodine removal function.

INSERT: 3.6-71-02

Valves in containment with remote position indication may be checked using remote position indicaiton.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6A.2

fan cooler units

Cooler

Operating each ~~[required]~~ containment cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the containment cooling train occurring between surveillances. It has also been shown to be acceptable through operating experience.

92

fan coolers are operated during normal plant operation,

fan cooler units

SR 3.6.6A.3

fan cooler

1400

Verifying that ~~each [required] containment cooling train ESW cooling~~ flow rate to each cooling unit is ≥ 700 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3). The Frequency was developed considering the known reliability of the Cooling Water System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

the service water

92 day

SR 3.6.6A.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 8). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

5

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6~~A~~.5 and SR 3.6.6~~A~~.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment ~~High-3~~ High-High pressure signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The ~~18~~ 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. ~~Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

~~The surveillance of containment sump isolation valves is also required by SR 3.5.2.5. A single surveillance may be used to satisfy both requirements.~~

SR 3.6.6~~A~~.7

This SR requires verification that each ~~[required]~~ containment ~~cooling train actuates~~ fan cooler unit starts and damper re-positions to the emergency mode upon receipt of an actual or simulated safety injection signal. The ~~18~~ 24 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. ~~See SR 3.6.6A.5 and SR 3.6.6A.6, above, for further discussion of the basis for the 18 month Frequency.~~

Insert: B 3.6-73-01

SR 3.6.6~~A~~.~~8~~ 9

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at ~~[the first refueling and at]~~ 10 year intervals is considered adequate to detect obstruction of the nozzles.

(continued)

**INSERT: 3.6-73-01**

SR 3.6.6.8

This SR verifies that the required Fan Cooler Unit testing is performed in accordance with Specification 5.5.10, Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP .

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A~~, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43~~.
2. 10 CFR 50, Appendix K.
3. FSAR, ~~Section [ ]~~ Sections 6.3 and 6.4.
4. FSAR, Section ~~[ ]~~ 14.3.
5. ~~FSAR, Section [ ].~~
6. ~~FSAR, Section [ ].~~
7. ~~FSAR, Section [ ].~~

5 ~~8.~~ ASME, Boiler and Pressure Vessel Code, Section XI.

Insert:
B 3.6-74-01

**INSERT: 3.6-73-01**

6. WCAP - 12269, Containment Margin Improvement Analysis for IP-3 Unit 3, Rev. 1.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.6:
# "Containment Spray System and Containment Fan Cooler System"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.6.6 - Containment Spray System and Containment Fan Cooler System

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.7: "Spray Additive System"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.6 CONTAINMENT SYSTEMS

### 3.6.7 Spray Additive System

LCO 3.6.7 The Spray Additive System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Spray Additive System inoperable. | A.1 Restore Spray Additive System to OPERABLE status. | 72 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3. | 6 hours |
| | <u>AND</u> | |
| | B.2 Be in MODE 5. | 84 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.7.1 | Verify each spray additive manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position. | 31 days |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.7.2 | Verify spray additive tank solution volume is ≥ 4000 gal. | 184 days |
| SR 3.6.7.3 | Verify spray additive tank NaOH solution concentration is ≥ 35% and ≤ 38% by weight. | 184 days |
| SR 3.6.7.4 | Verify each spray additive automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. | 24 months |
| SR 3.6.7.5 | Verify spray additive system flow from each flow path. | 5 years |

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.7 Spray Additive System

BASES

### BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA).

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, sodium hydroxide (NaOH) is the preferred spray additive. The NaOH added to the spray also ensures an alkaline pH of the solution recirculated from the containment sump. This pH band minimizes the evolution of iodine as well as the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

The Spray Additive System consists of one spray additive tank that is shared by the two trains of containment spray. Each train provides a flow path from the spray tank to a containment spray pump and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line. The eductors are designed to ensure that the pH of the spray mixture is between 9.0 and 10.0.

The Containment Spray System actuation signal opens the valves from the spray additive tank to the spray pump suctions after a 2 minute delay. The 35% to 38% NaOH solution is drawn into the spray pump suctions. The spray additive tank capacity provides for the addition of NaOH solution to all of the water sprayed from the RWST into containment via the Containment Spray System.

BASES

BACKGROUND (Continued)

The percent solution and volume of solution sprayed into containment ensures a long term equilibrium containment sump pH of approximately 9.0. This ensures the continued iodine retention effectiveness of the sump water during the recirculation phase of spray operation and also minimizes the occurrence of chloride induced stress corrosion cracking of the stainless steel recirculation piping.

APPLICABLE SAFETY ANALYSES

The Spray Additive System, in conjunction with the Fan Cooler System, is essential to the removal of airborne iodine within containment following a DBA.

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value volume following the accident. The analysis assumes that 100% of containment is covered by the spray (Ref. 1).

The DBA response time assumed for the Spray Additive System is the same as for the Containment Spray System (plus a 2 minute delay) and is discussed in the Bases for LCO 3.6.6, "Containment Spray and Fan Cooler System."

The DBA analyses assume that one train of the Containment Spray System is inoperable and that the spray additive is added to the remaining Containment Spray System flow path.

The Spray Additive System satisfies Criterion 3 of 10 CFR 50.36.

LCO

The Spray Additive System reduces the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to provide NaOH injection into the spray flow until the Containment Spray System suction path is switched from the RWST to the recirculation sump or

BASES

LCO (continued)

containment sump, and to raise the average spray solution pH to a level conducive to iodine removal, namely, to between 7.9 and 10.0. This pH range maximizes the effectiveness of the iodine removal mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

If the Spray Additive System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System and Containment Fan Cooler System are available and would remove iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

BASES

ACTIONS (continued)

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.2 (continued)

injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.5

To ensure that the correct pH level is established in the borated water solution provided by the Containment Spray System, flow in the Spray Additive System is verified once every 5 years. This SR provides assurance that NaOH will be introduced into the flow path upon Containment Spray System initiation. This test is satisfied by the Inservice Test Program verification of the spray additive tank check valve. Water may be used in lieu of NaOH for the performance of this SR which is not intended to require transfer of NaOH. Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow.

REFERENCES

1. FSAR, Chapters 6 and 14.

## Indian Point 3
## Improved Technical Specifications (ITS)
## Conversion Package

# Technical Specification 3.6.7:
# "Spray Additive System"

# PART 2:

# CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.3-5a | 179 | 179 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-6 | 145 | 145 | No TSCRs | No TSCRs for this Page | N/A |
| T 4.1-2(1) | 139 | 139 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-2 | 172 TSCR 98-043 | 172 TSCR 98-043 | IPN 98-043 | Instrument Channel Surveillance Intervals Extended to 24 Months | Incorporated |
| 4.5-7 | 178 | 178 | No TSCRs | No TSCRs for this Page | N/A |

2) RCS temperature and the source range detectors are monitored hourly;

and

3) no operations are permitted which would reduce the boron concentration of the reactor coolant system.

SEE ITS 3.4.12

8. When the RCS average cold leg temperature ($T_{cold}$) is below 319°F, or when RHR is in service (i.e., not isolated from the RCS), no safety injection pumps shall be energized and aligned to feed the RCS.

9. The requirements of 3.3.A.8 may be relaxed to allow one safety injection pump energized and aligned to feed the RCS under the following circumstances:

a. emergency boration; OR

b. for pump testing, for a period not to exceed 8 hours; OR

c. loss of RHR cooling.

10. The requirements of 3.3.A.8 may be further relaxed when the RCS is < 200°F, such that two safety injection pumps may be energized and aligned to feed the RCS under the following circumstances:

a. the RCS is vented with an opening greater than or equal to the size of one code pressurizer safety valve flange, OR

b.. indicated pressurizer level is at 0% and the plant is vented in accordance with Technical Specification 3.1.A.8.c.1. (Alternate methods and instrumentation may be used to confirm actual RCS elevation.)

B. <u>Containment Cooling and Iodine Removal Systems</u>

LCO 3.6.7 Applicability

Mode 1, 2, 3 and 4 A.3

~~1.~~ ~~The reactor shall not be brought above the cold shutdown condition unless the following~~ requirements are met:

SR 3.6.7.2

SR 3.6.7.3

Verify every 184 days M.4

a. The spray additive tank contains a minimum of 4000 gallons of solution with a sodium hydroxide concentration ≥35% and ≤38% by weight.

SEE ITS 3.6.6

b. The five fan cooler-charcoal filter units and the two spray pumps, with their associated valves and piping, are operable.

2. The requirements of 3.3.B.1 may be modified to allow any one of the following components to be inoperable at one time:

SEE ITS 3.6.6

a. Fan cooler unit 32, 34, or 35 or the flow path for fan cooler unit 32, 34, or 35 may be out of service for a period not to exceed 24 hours provided both containment spray pumps are operable.

OR

Fan cooler unit 31 or 33, or the flow path for fan cooler unit 31 or 33 may be out of service for a period not to exceed 7 days provided both containment spray pumps are operable.

b. One containment spray pump may be out of service for a period not to exceed 24 hours, provided the five fan cooler units are operable.

c. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided it is restored to an operable status within 24 hours and all valves in the system that provide the duplicate function are operable.

LCO 3.6.7

~~3.~~ If the Containment Cooling and Iodine Removal are not restored to meet the requirements of 3.3.B.1 within the time period specified in 3.3.B.2, then:

Req. Act B.1 B.2

a. If the reactor is critical, it shall be in the hot shutdown condition within ~~four~~ 6 hours and in the cold shutdown condition within ~~the following 24~~ 84 hours. L.2

b. ~~If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.B.1 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.~~ M.3

Add Condition A and associated Req. Actions L.1



Amendment No. ~~34~~, ~~132~~, 145

TABLE 4.1-2 (Sheet 1 of 2)

| FREQUENCIES FOR SAMPLING TESTS | | | |
|---|---|---|---|
| Sample | Analysis | Frequency | Maximum Time Between Analysis |
| 1. Reactor Coolant | Gross Activity(1)<br>Tritium Activity<br>Boron concentration<br>Radiochemical (gamma)(2) Spectral Check<br>Oxygen and Chlorides Concentration<br>Fluorides Concentration<br>$\bar{E}$ Determination (3)<br>Isotopic Analysis for I-131, I-133, I-135 | 5 days/week(1)(4)<br>Weekly(1)<br>2 days/week<br>Monthly<br>3 times per 7 days<br>Weekly<br>Semi-Annually<br>Once per 14 days(3) | 3 days(4)<br>10 days<br>5 days<br>45 days<br>3 days<br>10 days<br>30 weeks<br>20 days |
| 2. Boric Acid Tank | Boron Concentration, Chlorides | Weekly | 10 days |
| 3. Spray Additive Tank | NaOH Concentration | ~~Monthly~~ | ~~45 days~~ 184 days L.3 |
| 4. Accumulators | Boron Concentration | Monthly | 45 days |
| 5. Refueling Water Storage Tank | Boron Concentration pH, Chlorides<br>Gross Activity | Monthly<br>Quarterly | 45 days<br>16 weeks |
| 6. Secondary Coolant | I-131 Equivalent (Isotopic Analysis)<br>Gross Activity | Monthly<br>3 times per 7 days | 45 days<br>3 days |
| 7. Component Cooling Water | Gross Activity, Corrosion Inhibitor and pH | Monthly | 45 days |
| 8. Spent Fuel Pool (when fuel stored) | Gross Activity Boron Concentration, Chlorides | Monthly | 45 days |

SEE CTS MASTER MARKUP

SR 3.6.7.3

SEE CTS MASTER MARKUP

Amendment No. 139

ITS 3.6.7

2. Containment Spray System

SR 3.6.7.4 ~~a.~~ System tests shall be performed at least once per 24 months. ~~The tests shall be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed.~~ (A.5) Operation of the system is initiated by ~~tripping the normal~~ actual or simulated (A.4) actuation instrumentation.

SEE ITS 3.6.6 b. The spray nozzles shall be checked for proper functioning at least every five years.

SR 3.6.7.4 c. ~~The tests will be considered satisfactory if visual observations indicate all components have operated satisfactorily.~~ (A.5)

SEE ITS 3.3.3

3. Containment Hydrogen Monitoring Systems

a. Containment hydrogen monitoring system tests shall be performed at intervals no greater than six months. The tests shall include drawing a sample from the fan cooler units.

b. The above tests will be considered satisfactory if visual observations and control panel indication indicate that all components have operated satisfactorily.

Add SR 3.6.7.1 (M.1)

Add SR 3.6.7.5 (M.2)

B. Component Tests

1. Pumps

SEE ITS 3.5.2 3.5.3 3.6.6

a. The safety injection pumps, residual heat removal pumps, containment spray pumps and the auxiliary component cooling water pumps shall be started at quarterly intervals. The recirculation pumps shall be started at least once per 24 months.

b. Acceptable levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least fifteen minutes.

2. Valves

<3.6.7.4>

a. Each spray additive valve shall be cycled ~~by operator action with the pumps shut down~~ at least once per 24 months. A.6

SEE ITS 3.5.1

b. The accumulator check valves shall be checked for operability at least once per 24 months.

SEE ITS 3.4.14

c. The following check valves shall be checked for gross leakage at least once per 24 months:

| | | | |
|---|---|---|---|
| 857A & G | 857J | 857S & T | 897B |
| 857B | 857K | 857U & W | 897C |
| 857C | 857L | 895A | 897D |
| 857D | 857M | 895B | 838A |
| 857E | 857N | 895C | 838B |
| 857F | 857P | 895D | 838C |
| 857H | 857Q & R | 897A | 838D |



Amendment No. ~~123~~, ~~129~~, ~~148~~, 178

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.7:
# "Spray Additive System"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.6.7 - Spray Additive System

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.3.B.1 specifies the Applicability for containment cooling and iodine removal systems (which includes the spray additive system) as whenever the reactor is above cold shutdown (i.e., Modes 1, 2, 3 and 4). ITS 3.6.7 maintains this Applicability by requiring that the spray additive system is Operable in Modes 1, 2, 3 and 4. This is an administrative change with no impact on safety because there is no change to the CTS Applicability.

A.4 CTS 4.5.A.2.a requires a functional test of the containment spray system (including the spray additive system) and specifies that "operation of the system is initiated by tripping the normal actuation instrumentation." ITS SR 3.6.7.4 maintains the requirement for a functional test of the spray additive system; however, the test may be initiated by either an actual or simulated signal. This change is acceptable because use of an actual instead of a simulated or "test" signal will not affect the performance of the test because the equipment being tested cannot discriminate between an actual and simulated signal. This is an administrative change with no impact on safety because the use of an actual or simulated signal does not change the validity of the test as a verification of plant response to the event.

A.5 CTS 4.5.A.2.a specifies requirements for a functional test of the containment spray system and CTS 4.5.A.2.c establishes the acceptance criteria for this test as "the tests will be considered satisfactory if visual observations indicate all components have operated satisfactorily." ITS SR 3.6.7.4 maintains the requirement for a functional test of the spray additive system; however, the statement that appropriate verification of system performance is limited to visual observations that all components have operated is deleted. This change is acceptable because this type of generic statement is generally not included in the acceptance criteria of either the CTS or ITS. This is an administrative change with no adverse impact on safety because there is no change to the existing requirements.

A.6 CTS 4.5.B.2.a specifies that each spray additive valve shall be cycled by operator action with the pumps shut down at least once per 24 months. ITS SR 3.6.7.4 requires verification every 24 months that each automatic valve in the Spray Additive System flow path actuates to its correct position. ITS SR 3.6.7.4 is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This is an administrative change with no adverse impact on safety because ITS SR 3.6.7.4 maintains the requirement to cycle every 24 months all of the valves that are required to function for the spray additive system to perform its safety function.

MORE RESTRICTIVE

M.1 ITS SR 3.6.7.1 is added to require verification every 31 days that each spray additive manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position. There is no equivalent requirement in the CTS. This change is needed because it provides assurance that the proper flow paths exist for the containment spray additive system operation. This change has no adverse impact on safety.

M.2 ITS SR 3.6.7.5 is added to require verification every five years of spray additive rate from each solution tank's flow path. There is no equivalent requirement in the CTS. The addition of this surveillance requirement is needed because it provides assurance that NaOH will flow into the flow path upon containment spray system initiation. This change has no adverse impact on safety.

M.3 CTS 3.3.B.3 establishes the Actions required if either containment spray (including the spray additive system) is not restored to meet CTS requirements within specified completion times. CTS 3.3.B.3.a specifies that, if the reactor is critical when requirements are not met, then the reactor must be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within the following 24 hours. However, if the reactor is subcritical when requirements are not met, CTS 3.3.B.3.b requires only that reactor coolant system temperature and pressure not be increased more than 25°F and 100 psi, respectively, over existing values with the requirement to proceed to cold shutdown (Mode 5) deferred by 48 hours.

Under the same conditions, ITS 3.6.7, Required Actions B.1 and B.2, require that the reactor be in Mode 3 in 6 hours (See ITS 3.6.7, DOC L.2) and in Mode 5 in 84 hours (See ITS 3.6.7, DOC L.2), regardless of the status of the unit when the Condition is identified. The allowance provided in CTS 3.3.B.3.b is deleted.

This change is needed to eliminate the ambiguity created by CTS 3.3.B.3.b when performing a reactor shutdown and cooldown required by CTS 3.3.B.3.a and to ensure that the plant is placed outside the LCO

Applicability promptly when the LCO requirements are not met. This change is acceptable because placing the plant outside the LCO Applicability when LCO requirements are not met is conservative and there is no change in the CTS 3.3.B.3 requirement. This change has no significant adverse impact on safety.

M.4 CTS 3.3.B.1.a requires that the spray additive tank solution volume be maintained ≥ 4000 gallons; however, there is no requirement for the periodic verification that this requirement is met. ITS SR 3.6.7.2 is added to require verification every 184 days that requirements for the minimum solution volume in the spray additive tank are met. This change is needed because it requires periodic verification that spray additive tank requirements are met. This Surveillance Frequency is acceptable because of the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Additionally, the tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected. This change has no significant adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.3.B.1.a establishes requirements for the spray additive system. If the requirements of CTS 3.3.B.1.a are not met, then CTS 3.3.B.3 requires initiation of a plant shutdown because no other allowable out of service time (AOT)is specified. Under the same conditions, ITS LCO 3.6.7, Required Action A.1, provides an AOT of 72 hours before a reactor shutdown is required. This change is acceptable because the containment spray system still provides significant capability to remove iodine from the containment atmosphere in the event of a design basis accident even if the spray additive system is degraded or completely unavailable. Additionally, the containment Fan Cooler Unit System is also available for iodine removal from the containment atmosphere following an accident. The 72-hour Completion Time takes in the redundant flow path capabilities and the low probability of the worst case design basis accident during this period. Therefore, this change has no adverse impact on safety.

L.2 CTS 3.3.B.3 establishes the Actions required if either containment spray (including the spray additive system) and/or containment fan cooler trains are not restored to meet CTS requirements within specified completion times. CTS 3.3.B.3.a specifies that, if the reactor is critical when requirements are not met, then the reactor must be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within the following 24 hours (See ITS 3.6.7, DOC M.3). Under the same conditions, ITS 3.6.7, Required Actions B.1 and B.2, require that the reactor be in Mode 3 in 6 hours and in Mode 5 in 84 hours.

This change is needed and is acceptable because placing the reactor in Mode 3 in 6 hours and in Mode 5 in 84 hours it allows additional time for attempting restoration of the containment spray additive system and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in Mode 3. Therefore, this change has no significant adverse impact on safety.

L.3 CTS 3.3.B.1.a and CTS Table 4.1-2, Item 3, requires verification that the NaOH concentration in the spray additive tank is within required limits every month with the maximum time between analyses of 45 days. ITS SR 3.6.7.3 maintains the requirement that the NaOH concentration in the spray additive tank is within required limits; however, the Surveillance Frequency is extended to 184 days with the maximum time between analyses determined by ITS SR 3.0.2 (i.e., 125% of normal SR Frequency). This relaxation in frequency is acceptable because the 184-day Frequency is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected. This change has no adverse impact on safety.

<u>REMOVED DETAIL</u>

None

## Indian Point 3
## Improved Technical Specifications (ITS)
## Conversion Package

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# Technical Specification 3.6.7:
# "Spray Additive System"

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## PART 4:

## No Significant Hazards Considerations
## for
## Changes between CTS and ITS
## that are
## Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.6.7 - Spray Additive System

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change revises the time one containment spray pump (and by extension the spray additive system) may be out of service. CTS 3.3.B.1.a establishes requirements for the spray additive system. If the requirements of CTS 3.3.B.1.a are not met, then CTS 3.3.B.3 requires initiation of a plant shutdown because no other allowable out of service time (AOT)is specified. Under the same conditions, ITS LCO 3.6.7, Required Action A.1, provides an AOT of 72 hours before a reactor shutdown is required.

   This change will not result in a significant increase in the probability of an accident previously evaluated because an inoperable spray additive system is not the precursor of any event. This change will not result in a significant increase in the consequences of an accident previously evaluated because the containment spray system still provides significant capability to remove iodine from the containment atmosphere in the event of a design basis accident even if the spray additive system is degraded or completely unavailable. Additionally, the containment Fan Cooler Unit System is also available for iodine removal from the containment atmosphere following an accident. The 72-hour Completion Time takes in the redundant flow path capabilities and the low probability of the worst case design basis accident during this period.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the containment spray system still provides significant capability to remove iodine from the containment atmosphere in the event of a design basis accident even if the spray additive system is degraded or completely unavailable. Additionally, the containment Fan Cooler Unit System is also available for iodine removal from the containment atmosphere following an accident. The 72-hour Completion Time takes in the redundant flow path capabilities and the low probability of the worst case design basis accident during this period.

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LESS RESTRICTIVE
<u>("L.2" Labeled Comments/Discussions)</u>

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   CTS 3.3.B.3 establishes the Actions required if either containment spray (and by extension the spray additive system) is not restored to meet CTS requirements within specified completion times. CTS 3.3.B.3.a specifies that, if the reactor is critical when requirements are not met, then the reactor must be in hot shutdown (Mode 3) within 4 hours and cold shutdown (Mode 5) within the following 24 hours (See ITS 3.6.7, DOC M.3). Under the same conditions, ITS 3.6.7, Required Actions B.1 and B.2, require that the reactor be in Mode 3 in 6 hours and in Mode 5 in 84 hours.

This change will not result in a significant increase in the probability of an accident previously evaluated because an inoperable spray additive system is not the precursor of any event. This change will not result in a significant increase in the consequences of an accident previously evaluated because the containment spray system still provides significant capability to remove iodine from the containment atmosphere in the event of a design basis accident even if the spray additive system is degraded or completely unavailable. Additionally, the containment Fan Cooler Unit System is also available for iodine removal from the containment atmosphere following an accident. The 72-hour Completion Time takes in the redundant flow path capabilities and the low probability of the worst case design basis accident during this period.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the containment spray system still provides significant capability to remove iodine from the containment atmosphere in the event of a design basis accident even if the spray additive system is degraded or completely unavailable. Additionally, the containment Fan Cooler Unit System is also available for iodine removal from the containment atmosphere following an accident. The 72-hour Completion Time takes in the redundant flow path capabilities and the low probability of the worst case design basis accident during this period.

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LESS RESTRICTIVE
("L.3" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change extends the surveillance interval for verification of the spray additive tank NaOH concentration. CTS 3.3.B.1.a and CTS Table 4.1-2, Item 3, requires verification that the NaOH concentration in the spray additive tank is within required limits every month with the maximum time between analyses of 45 days. ITS SR 3.6.7.3 maintains the requirement that the NaOH concentration in the spray additive tank is within required limits; however, the Surveillance Frequency is extended to 184 days with the maximum time between analyses determined by ITS SR 3.0.2 (i.e., 125% of normal SR Frequency).

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because the 184-day Frequency is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because the spray additive system is a mitigating system that assists in reducing the iodine fission products inventory in the containment atmosphere resulting from a design basis accident. During the period the spray additive system is inoperable the containment spray system would still be available and would remove some iodine from the containment atmosphere in the event of a design basis accident. Also, the containment cooling and air filtration units would be available for iodine removal. The surveillance interval is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184-day Frequency is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.7:
## "Spray Additive System"

## PART 5:

## NUREG-1431
## Annotated to show differences between NUREG-1431 and ITS

**Status of NUREG 1431 Generic Changes for ITS 3.6.7**

This ITS Specification is based on NUREG-1431 Specification No. **3.6.7** as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| N/A | N/A | NO GENERIC CHANGES ARE POSTED AGAINST THIS SPECIFICATION. | Not Applicable | Not Applicable | N/A |

<CTS>

## 3.6 CONTAINMENT SYSTEMS

### 3.6.7 Spray Additive System ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~

<3.3.B.1.a>

LCO 3.6.7 The Spray Additive System shall be OPERABLE.

<3.3.B.1>
<DOC A.3>

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <DOC L.1> | A. Spray Additive System inoperable. | A.1 Restore Spray Additive System to OPERABLE status. | 72 hours |
| <3.3.B.3.a> <DOC L.2> <DOC M.3> | B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br>AND<br>B.2 Be in MODE 5. | 6 hours<br><br>84 hours |

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| <DOC M.1> | SR 3.6.7.1 Verify each spray additive manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position. | 31 days |

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

| | SURVEILLANCE | | FREQUENCY |
|---|---|---|---|
| <3.3.B.1.a> <DOC M.4> | SR 3.6.7.2 | Verify spray additive tank solution volume is ≥ [2568] 4000 gal ~~and ≤ [4000] gal~~. | 184 days |
| <3.3.B.1.a> <Table 4.1-2, #3> <DOC L.3> | SR 3.6.7.3 | Verify spray additive tank [NaOH] solution concentration is ≥ [30] 35% and ≤ [32] 38% by weight. | 184 days |
| <4.5.A.2.a> <4.5.A.2.c> <DOC A.4> <4.5.B.2.a> <DOC A.5> <DOC A.6> | SR 3.6.7.4 | Verify each spray additive automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. | [18] 24 months |
| <DOC M2> | SR 3.6.7.5 | Verify spray additive system flow ~~[rate]~~ from each ~~solution's~~ flow path. | 5 years |

OT

## B 3.6 CONTAINMENT SYSTEMS

## B 3.6.7 Spray Additive System ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~

## BASES

**BACKGROUND**

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA).

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, sodium hydroxide (NaOH) is the preferred spray additive. The NaOH added to the spray also ensures ~~a pH value of between 8.5 and 11.0~~ an alkaline pH of the solution recirculated from the containment sump. This pH band minimizes the evolution of iodine as well as the occurrence of chloride ~~and caustic~~ stress corrosion on mechanical systems and components.

~~Eductor Feed Systems Only~~

The Spray Additive System consists of one containment spray additive tank that is shared by the two trains of spray additive equipment. Each train ~~of equipment~~ provides a flow path from the spray additive tank to a containment spray pump and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line. The eductors are designed to ensure that the pH of the spray mixture is between ~~8.5 and 11.0~~ 9.0 and 10.0

(continued)



Typical

BASES

BACKGROUND (continued)

~~Gravity Feed Systems Only~~

~~The Spray Additive System consists of one spray additive tank, two parallel redundant motor operated valves in the line between the additive tank and the refueling water storage tank (RWST), instrumentation, and recirculation pumps. The NaOH solution is added to the spray water by a balanced gravity feed from the additive tank through the connecting piping into a weir within the RWST. There, it mixes with the borated water flowing to the spray pump suction. Because of the hydrostatic balance between the two tanks, the flow rate of the NaOH is controlled by the volume per foot of height ratio of the two tanks. This ensures a spray mixture pH that is ≥ 8.5 and ≤ 11.0.~~

DB.1

The Containment Spray System actuation signal opens the valves from the spray additive tank to the spray pump suctions ~~or the containment spray pump start signal opens the valves from the spray additive tank~~ after a 5 minute delay. The ~~28% to 31%~~ 35% to 38% NaOH solution is drawn into the spray pump suctions. The spray additive tank capacity provides for the addition of NaOH solution to all of the water sprayed from the RWST into containment, via the containment spray system. The percent solution and volume of solution sprayed into containment ensures a long term equilibrium containment sump pH of ~~≥ 9.0 and ≤ 9.5~~ approximately 9.0. This ensures the continued iodine retention effectiveness of the sump water during the recirculation phase of spray operation and also minimizes the occurrence of chloride induced stress corrosion cracking of the stainless steel recirculation piping.

2

APPLICABLE SAFETY ANALYSES

The Spray Additive System, in conjunction with the Air Cooler System, is essential to the removal of airborne iodine within containment following a DBA.

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value volume following the accident. The analysis assumes that 100% of containment is covered by the spray (Ref. 1).

The DBA response time assumed for the Spray Additive System (plus a 2 minute delay) is the same as for the Containment Spray System and is discussed in the Bases for LCO 3.6.6, "Containment Spray and ~~Cooling Systems~~ Fan Cooler."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The DBA analyses assume that one train of the Containment Spray System/~~Spray Additive System~~ is inoperable and that the ~~entire~~ spray additive ~~tank volume~~ is added to the remaining Containment Spray System flow path.

The Spray Additive System satisfies Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36

LCO

The Spray Additive System ~~is necessary to~~ reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to provide NaOH injection into the spray flow until the Containment Spray System suction path is switched from the RWST to the containment sump, and to raise the average spray solution pH to a level conducive to iodine removal, namely, to between ~~7.8 and 11.0~~. This pH range maximizes the effectiveness of the iodine removal mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

or recirculation sump

7.9 and 10.0

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

If the Spray Additive System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of

(continued)

BASES

ACTIONS

A.1 (continued)

the Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System ~~would still be~~ and Containment Fan Cooler System are available and would remove ~~some~~ iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

24

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.5

Insert: B3.6-114

To ensure that the correct pH level is established in the borated water solution provided by the Containment Spray System, ~~the~~ flow ~~rate~~ in the Spray Additive System is verified once every 5 years. This SR provides assurance that ~~the correct amount of~~ NaOH will be metered introduced into the flow path upon Containment Spray System initiation. Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow ~~rate~~.

REFERENCES

1. FSAR, Chapter [15]. Chapters 6 and 14

NUREG-1431 Markup Inserts
ITS SECTION 3.6.7 - Spray Additive System

INSERT: 3.6-114-01

This test is satisfied by the Inservice Test Program verification of the spray additive tank check valve. Water may be used in lieu of NaOH for the performance of this SR which is not intended to require the transfer of NaOH.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.7:
# "Spray Additive System"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.6.7 - Spray Additive System

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.8:
## "Hydrogen Recombiners"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.6 CONTAINMENT SYSTEMS

### 3.6.8 Hydrogen Recombiners

LCO 3.6.8 Two hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One hydrogen recombiner inoperable. | A.1 | --------NOTE---------<br>LCO 3.0.4 is not applicable.<br>---------------------<br><br>Restore hydrogen recombiner to OPERABLE status. | <br><br><br>30 days |
| B. Required Action and associated Completion Time not met. | B.1 | Be in MODE 3. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.8.1 | Perform a system functional test for each hydrogen recombiner. | 6 months |
| SR 3.6.8.2 | Visually examine each hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions. | 24 months |
| SR 3.6.8.3 | Perform a resistance to ground test for each heater phase. | 24 months |

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Recombiners

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) or steam line break (SLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiner systems are provided. Each consists of controls located in the control room, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.1 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus, and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.1 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are not exceeded. The limiting DBA relative to

BASES

APPLICABLE SAFETY ANALYSES (continued)

hydrogen generation is a LOCA. Hydrogen may accumulate in containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 2.0 v/o about 5 days after the LOCA and 3.0 v/o about 10 days later if no recombiner was functioning (Ref. 3). Initiating the hydrogen recombiners when the primary containment hydrogen concentration reaches 3.0 v/o will maintain the hydrogen concentration in the primary containment below flammability limits.

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 3).

The hydrogen recombiners satisfy Criterion 3 of 10 CFR 50.36.

BASES

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA, assuming a worst case single failure.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations in these MODES. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur)

BASES

ACTIONS

A.1 (continued)

for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note that states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

B.1

If the inoperable hydrogen recombiner cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum power for approximately 2 minutes and power is verified to be ≥ 60 kW.

Operating experience has shown that these components usually pass the Surveillance when performed at the 6 month Frequency.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1 (continued)

Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.8.2

This SR ensures there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failure involves loss of power, blockage of the internal flow, missile impact, etc.

A visual inspection is sufficient to determine abnormal conditions (e.g., loose wiring or structural connections, deposits of foreign materials, etc.) that could cause such failures. The 24 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.8.3

This SR requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms.

The 24 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES


1. 10 CFR 50.44.

2. 10 CFR 50, Appendix A.

3. FSAR Section 6.8.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.8:
## "Hydrogen Recombiners"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

### Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.3-12 | 115 | 115 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-13 | 115 | 115 | No TSCRs | No TSCRs for this Page | N/A |
| 3.3-19 | 145 IPN 97-175 | 145 IPN 97-175 | IPN 97-175 | Changes to Bases Pages | |
| 3.3-20 | 145 | 145 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-6 | 130 | 130 | No TSCRs | No TSCRs for this Page | N/A |
| 4.5-10 | 148 | 148 | No TSCRs | No TSCRs for this Page | N/A |

SEE
CTS RELOCATED

**AND**

1. Within 72 hours after identification of the inoperability of both installed monitoring channels, restore one monitoring channel to operable status.

**OR**

2. Submit a Special Report to the NRC pursuant to Technical Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the monitoring systems.

I. Electric Hydrogen Recombiner System

LCO 3.6.8
Applicability

1. Two independent Hydrogen Recombiner Systems shall be OPERABLE ~~whenever the reactor $T_{avg}$ exceeds 350°F~~. Mode 1 and 2 L.1

Req. Act A.1
Req. Act B.1

~~a.~~ With one Hydrogen Recombiner System inoperable, restore the inoperable system to operable status within 30 days, or be in the HOT SHUTDOWN CONDITION Mode 3 within the next 6 hours ~~and subsequently reduce $T_{avg}$ to less than or equal to 350°F within the following 30 hours.~~ L.1

Note to Req. Act A.1

~~b.~~ The reactor operating condition may be escalated while one Hydrogen Recombiner System is inoperable provided the requirements of section 3.3.I.1.a. above, are satisfied.

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A.1

(A.1)



A total of six service water pumps are installed. Only two of the set of three service water pumps on the header designated the essential header are required immediately following a postulated loss-of-coolant accident.[8] During the recirculation phase of the accident, two service water pumps on the non-essential header will be manually started to supply cooling water for one component cooling system heat exchanger, one control room air conditioner, and one diesel generator; the other component cooling system heat exchanger, the other control room air conditioner, the two other diesel generators and remaining safety related equipment are cooled by the essential service water header.[14] During the recirculation phase of the accident, both control room air conditioner units may be cooled by the essential service water header.

The operability requirements on service water temperature monitoring instrumentation and the frequency of service water temperature monitoring insures that appropriate action can be taken to preclude operation beyond established limits. The locations selected for monitoring river water temperature are typically at the circulating or service water inlets, at the circulating water inlet boxes to the condenser hotwells or at the service water supply header to the fan cooler units. Temperature measurements at each of these locations are representative of the river water temperature supplied to cool plant heat loads. Alternate locations may be acceptable on this basis. The limit on the service water maximum inlet temperature insures that the service water and component cooling water systems will be able to dissipate the heat loads generated in the limiting design basis accident[15]. This restriction allows up to seven hours for river water temperature transients which may temporarily increase the service water inlet temperature due to tidal effects to dissipate.

The operability of the equipment and systems required for the control of hydrogen gas ensures that this equipment is available to maintain the hydrogen concentration within containment below the flammable limit during post-LOCA conditions. Hydrogen concentration exceeding the flammable limit could potentially result in a containment wide hydrogen burn. This could lead to overpressurization of containment, a breach of CONTAINMENT INTEGRITY, containment leakage, unacceptably high offsite doses, and damage to safety-related equipment located in containment. Two full rated recombiner units are provided in order to control the hydrogen evolved in containment following a loss-of-coolant accident. Each unit is capable of preventing the hydrogen concentration from exceeding the flammable limit. Each recombiner is installed such that independence is maintained and redundancy is assured. Each hydrogen recombiner system consists of a recombiner located inside containment, and a separate power supply, and control panel located outside containment such that they are accessible following a design basis accident.





TSCR 97-175

A.1

The containment hydrogen monitoring system consists of two safety related hydrogen concentration measurement cabinets with sample lines which pass through the containment penetrations to each containment fan cooler unit plenum. Two of the five sampling lines (from containment fan cooler units nos. 32 and 35) are routed to a common source line and then to a hydrogen monitor. The other three sample lines (from containment fan cooler units nos. 31, 33 and 34) are likewise headered and routed to the other hydrogen monitor. Each monitor has a separate return line. The design hydrogen concentration for operating the recombiner is established at 3% by volume. Conservative calculations indicate that the hydrogen content within the containment will not reach 3% by volume until 10 days after a loss-of-coolant accident.[10] There is, therefore, no need for immediate operation of the recombiner following an accident.

Auxiliary Component Cooling Pumps are provided to deliver cooling water for the two Recirculation Pumps located inside the containment. Each recirculation pump is fed by two Auxiliary Component Cooling Pumps. A single Auxiliary Component Cooling Pump is capable of supplying the necessary cooling water required for a recirculation pump during the recirculation phase following a loss-of-coolant accident.

The control room ventilation is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage.

Radiation monitor R-1 is not part of the Control Room Ventilation System. NRC letter dated January 27, 1982 concluded that, at IP3, "radiation monitors for makeup air are not required." NYPA has also demonstrated (calculation dated May 29, 1992) that Central Control Room (CCR) isolation is not required for maintaining radiation exposure within General Design Criteria 19 limits following a fuel handling accident or gas-decay-tank rupture. For a loss of coolant accident, CCR isolation is initiated by the safety injection signal.

The control room is equipped with two independent toxic gas monitoring systems. One system in the control room consists of a channel for oxygen (with two oxygen detectors) and a channel each for ammonia and chlorine. The second system in the control room ventilation intake consists of one channel each for oxygen, ammonia and chlorine. Oxygen detectors are used to indirectly monitor changes in carbon dioxide levels.

SEE ITS 5.5.10

(3) A locally generated DOP test of the HEPA filters at ± 20% of the accident design flow rate and ambient conditions shall show ≥ 99% DOP removal.

(4) Visual inspection in accordance with the applicable sections of ANSI N 510 (1975) of filter installations.

7. Electric Hydrogen Recombiner Systems

a. Each hydrogen recombiner system shall be demonstrated OPERABLE:

SR 3.6.8.1

~~1)~~ At least once every 6 months by verifying, during a Hydrogen Recombiner System Functional test, ~~that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 kW, and~~ LA.1

~~2)~~ At least once per 24 months by:

~~a) Performing CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,~~ LA.2

SR 3.6.8.2

~~b)~~ Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure ~~(i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and~~ LA.1

SR 3.6.8.3

c) Verifying the integrity of all heater electrical circuits by performing a resistance to ground test ~~following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.~~ LA.1

Amendment No. ~~79~~, ~~123~~, 130

A.1

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to similarly prevent clogging of these adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radio-iodine by control room personnel. The in-place test results should indicate a system leak tightness of less than or equal to one percent leakage for the charcoal adsorbers and a HEPA filter efficiency of greater than or equal to 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a methyl iodide removal efficiency of greater than or equal to 90 percent for expected accident conditions.

With the efficiencies of the HEPA filters and charcoal adsorbers as specified, further assurance is provided that the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10CFR Part 50.

A pressure drop across the combined HEPA filters and charcoal adsorbers of less than or equal to 6.0 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability. Proper operation of the system fans should also be verified at least every refueling by either direct or indirect measurements.

If results of charcoal tests are unsatisfactory, two additional samples may be tested. If both of these tests are acceptable, the charcoal may be considered satisfactory for use in the plant. Should the charcoal of any of these air filtration systems fail to satisfy the test criteria outlined in this specification, the charcoal beds will be replaced with new charcoal which satisfies the requirements for new charcoal outlined in Regulatory Guide 1.52 (Revision June, 1973).

The hydrogen recombiner system is an engineered safety feature which would be used only following a loss-of-coolant accident to control the hydrogen evolved in the containment. The system is not expected to be needed until approximately 10 days have elapsed following the accident. At this time, the hydrogen concentration in the containment will have reached 3.0% by volume, which is the design concentration for starting the recombiner system.(3) Actual starting of the system will be based upon containment atmosphere sample analysis. The required surveillance testing of each unit will demonstrate the operability of the system. The bi-annual testing of the containment hydrogen monitoring system will demonstrate the availability of this system.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.8:
## "Hydrogen Recombiners"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.6.8 - Hydrogen Recombiners

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

MORE RESTRICTIVE

None

DISCUSSION OF CHANGES
ITS SECTION 3.6.8 - Hydrogen Recombiners

LESS RESTRICTIVE

L.1 CTS 3.3.I.1 requires that two independent hydrogen recombiner systems are operable whenever reactor Tavg exceeds 350°F (i.e., Modes 1, 2, and 3). ITS LCO 3.6.8 maintains the requirement that two independent hydrogen recombiner systems must be operable; however, this requirement is applicable only in Modes 1 and 2. In conjunction with this change, the actions for one or more inoperable hydrogen recombiners are changed to require only that the plant be placed in Mode 3 rather than reduce reactor Tavg to less than 350°F (i.e., Mode 4). This change, eliminating the requirement for hydrogen recombiner Operability in Mode 3, is acceptable because in Mode 3 the hydrogen production rate and the total hydrogen produced after a LOCA would be significantly less than that calculated for the design basis accident LOCA; therefore, hydrogen recombiners are not needed to maintain hydrogen concentration in containment to less than the flammability limit. Therefore, this change has no adverse impact on safety.

REMOVED DETAIL

LA.1 CTS 4.5.7.a.1 and CTS 4.5.7.2.b require periodic inspections and performance of a functional test of the hydrogen recombiners and include detailed acceptance criteria. ITS SR 3.6.8.1 maintains the requirement for periodic performance of a functional test of the hydrogen recombiners; however, the detailed acceptance criteria are relocated to the Bases for ITS SR 3.6.8.1.

Additionally, CTS 4.5.7.a.2.c requires periodic performance of a resistance to ground check of the hydrogen recombiners and includes detailed acceptance criteria. ITS SR 3.6.8.3 maintains the requirement for periodic performance of a resistance to ground check of the hydrogen recombiners; however, the detailed acceptance criteria are relocated to the Bases for ITS SR 3.6.8.3.

This change is acceptable because LCO 3.6.8 still requires that hydrogen recombiners are Operable and ITS SR 3.6.8.1 and ITS SR 3.6.8.3 still require periodic performance of tests designed to verify Operability.

Therefore, the requirement to have hydrogen recombiners Operable is not changed. Therefore, this acceptance criteria which are design information can be adequately defined and controlled in the ITS Bases which require change control in accordance with ITS 5.5.12, Bases Control Program. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the requirement to maintain the hydrogen recombiner Operability. This change is a less restrictive administrative change with no impact on safety.

LA.2 CTS 4.5.7.a.2.a requires performance of a channel calibration of all recombiners instrumentation and control circuits every 24 months. This requirement is not included in ITS 3.6.8, and is relocated to the Final Safety Analysis Report (FSAR) and implemented by plant procedures. This requirement can be relocated to the FSAR because the requirement that recombiners are Operable is included in LCO 3.6.8, and that operability is verified by the performance of the surveillance requirements.

This change is a less restrictive administrative change with no impact on safety because ITS 3.6.8 maintains the requirements to have hydrogen recombiners Operable and maintains the requirements to perform periodic verification that demonstrates hydrogen recombiner Operability. Therefore, requirements to calibrate hydrogen recombiner instruments can be maintained in the FSAR with no significant adverse impact on safety.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.6.8:
# "Hydrogen Recombiners"

## PART 4:

## No Significant Hazards Considerations for Changes between CTS and ITS that are Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.6.8 - Hydrogen Recombiners

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change, eliminating the requirement for hydrogen recombiner Operability in Mode 3, will not result in a significant increase in the probability or consequences of an accident previously evaluated because in Mode 3 the hydrogen production rate and the total hydrogen produced after a LOCA would be significantly less than that calculated for the design basis accident LOCA; therefore, hydrogen recombiners are not needed to maintain hydrogen concentration in containment to less than the flammability limit.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because in Mode 3 the hydrogen production rate and the total hydrogen produced after a LOCA would be significantly less than that calculated for the design basis accident LOCA; therefore, hydrogen recombiners are not needed to maintain hydrogen concentration in containment to less than the flammability limit.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.8:
## "Hydrogen Recombiners"

## PART 5:

## NUREG-1431
## Annotated to show differences between
## NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.6.8

This ITS Specification is based on NUREG-1431 Specification No. **3.6.8**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| N/A | **N/A** | NO GENERIC CHANGES ARE POSTED AGAINST THIS SPECIFICATION. | Not Applicable | Not Applicable | **N/A** |

## 3.6 CONTAINMENT SYSTEMS

### 3.6.8 Hydrogen Recombiners ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual) (if permanently installed)~~

<3.3.I.1> LCO 3.6.8 Two hydrogen recombiners shall be OPERABLE.

<3.3.I.1>
<DOC L.1> APPLICABILITY: MODES 1 and 2.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| <3.3.I.1.b> <3.3.I.1.a> A. One hydrogen recombiner inoperable. | A.1 --------NOTE--------- LCO 3.0.4 is not applicable. -------------------- Restore hydrogen recombiner to OPERABLE status. | 30 days |
| B. Two hydrogen recombiners inoperable. | B.1 Verify by administrative means that the hydrogen control function is maintained. <br> AND <br> B.2 Restore one hydrogen recombiner to OPERABLE status. | 1 hour <br> AND <br> Once per 12 hours thereafter <br> 7 days |
| <3.3.I.1.a> <DOC L.1> ~~C~~ B. Required Action and associated Completion Time not met. | ~~C~~ B.1 Be in MODE 3. | 6 hours |

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | | FREQUENCY |
|---|---|---|---|
| <4.5.7.a.1> <DOC LA.1> | SR 3.6.8.1 | Perform a system functional test for each hydrogen recombiner. | [18] months 6 |
| <4.5.7.a.2.b> | SR 3.6.8.2 | Visually examine each hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions. | [~~18~~] months 24 |
| <4.5.7.a.2.c> <DOC LA.1> | SR 3.6.8.3 | Perform a resistance to ground test for each heater phase. | [~~18~~] months 24 |

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Recombiners ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual) (if permanently installed)~~

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) or steam line break (SLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiner systems are provided. Each consists of controls located in the control room, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. ~~The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner.~~ A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.1 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus, and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.1 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA.

(continued)

BASES

**APPLICABLE SAFETY ANALYSES (continued)**

Hydrogen may accumulate in containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

5

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 3.5 v/o about 6 days after the LOCA and 4.0 v/o about 2 days later if no recombiner was functioning (Ref. 3). Initiating the hydrogen recombiners when the primary containment hydrogen concentration reaches 3.5 v/o will maintain the hydrogen concentration in the primary containment below flammability limits.

2.0 10 3.0

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 4). ~~The Hydrogen Purge System is similarly designed such that one of two redundant trains is an adequate backup to the redundant hydrogen recombiners.~~

3

The hydrogen recombiners satisfy Criterion 3 of ~~the NRC Policy Statement~~.

10 CFR 50.36

(continued)

BASES (continued)

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA, assuming a worst case single failure.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations in these MODES. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

(continued)

BASES

ACTIONS

A.1 (continued)

Required Action A.1 has been modified by a Note that states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

~~B.1 and B.2~~

~~[Reviewer's Note: This Condition is only allowed for units with an alternate hydrogen control system acceptable to the technical staff.]~~

~~With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by [the containment Hydrogen Purge System/hydrogen recombiner/Hydrogen Ignitor System/Hydrogen Mixing System/Containment Air Dilution System/Containment Inerting System]. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. [Reviewer's Note: The following is to be used if a non-Technical Specification alternate hydrogen control function is used to justify this Condition: In addition, the alternate hydrogen control system capability must be verified once per 12 hours thereafter to ensure its continued availability.] [Both] the [initial] verification [and all subsequent verifications] may be performed as an administrative check by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is~~

(continued)

BASES

ACTIONS

~~B.1 and B.2 (continued)~~

~~maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.~~

B ~~C~~.1

If the inoperable hydrogen recombiner~~(s)~~ cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum power for approximately 2 minutes and power is verified to be ≥ 60 kW.

Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. 6

SR 3.6.8.2

This SR ensures there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failure involves loss of power, blockage of the internal flow, missile impact, etc.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.8.2 (continued)

A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The 18 24 month Frequency for this SR was developed considering ~~the~~ incidence of hydrogen recombiners failing the SR in the past is low.

Insert B3.6-120-01

SR 3.6.8.3

This SR requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms.

The 18 24 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A~~, GDC 41~~.
3. ~~Regulatory Guide 1.7, Revision [1].~~
3 ~~4~~. FSAR Section 15 6.8

NUREG-1431 Markup Inserts
ITS SECTION 3.6.8 - Hydrogen Recombiners

INSERT: 3.6-120-01

(e.g., loose wiring or structural connections, deposits of foreign materials, etc.)

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.8:
# "Hydrogen Recombiners"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.6.8 - Hydrogen Recombiners

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

None

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. A detailed description of the design, accident analysis assumptions, and Operability requirements are incorporated into the IP3 ITS Bases. These changes maintain the IP3 current licensing basis except as identified and justified in the CTS/ITS discussion of changes.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.9:
# "Isolation Valve Seal Water (IVSW) System"

## PART 1:

## Indian Point 3
## Improved Technical Specifications and Bases

## 3.6 CONTAINMENT SYSTEMS

### 3.6.9 Isolation Valve Seal Water (IVSW) System

LCO 3.6.9 The IVSW System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One IVSW system header inoperable.<br><br>OR<br><br>One IVSW automatic actuation valve inoperable. | A.1 Restore IVSW system to OPERABLE status. | 7 days |
| B. IVSW system inoperable for reasons other than Condition A. | B.1 Restore IVSW System to OPERABLE Status. | 24 hours |
| C. Required Action and associated Completion Time not met. | C.1 Be in MODE 3.<br><br>AND<br><br>C.2 Be in MODE 5. | 6 hours<br><br><br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.9.1 | Verify IVSW tank pressure is ≥ 47 psig. | 24 hours |
| SR 3.6.9.2 | Verify IVSW nitrogen supply bank includes a minimum of 3 cylinders and that each cylinder has a pressure ≥ 150 psig. | 24 hours |
| SR 3.6.9.3 | Verify the IVSW tank water volume is ≥ 144 gallons. | 24 hours |
| SR 3.6.9.4 | Verify the opening time of each air operated header injection valve is within limits. | In accordance with the Inservice Testing Program |
| SR 3.6.9.5 | Verify each automatic valve in the IVSW System actuates to the correct position on an actual or simulated actuation signal. | 24 months |
| SR 3.6.9.6 | Verify the leakage rate of water from the Isolation Valve Seal Water System in accordance with the Containment Leakage Rate Testing Program. | In accordance with the Containment Leakage Rate Testing Program. |

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Isolation Valve Seal Water (IVSW) System

BASES

BACKGROUND

The Isolation Valve Seal Water (IVSW) System improves the effectiveness of certain containment isolation valves (CIVs) by providing a water seal to valve leakage paths. This is accomplished by injecting water between the seats and stem packing of globe and double-disk type isolation valves and into the piping between other closed containment isolation valves. IVSW sealing water is maintained in a seal water supply tank filled with water and pressurized with nitrogen. The IVSW System is actuated in conjunction with automatic initiation of containment isolation and is applied to CIVs in lines connected to the Reactor Coolant System or exposed to the containment atmosphere during an accident. The seal water is injected at a pressure of at least 47 psig which is > 1.1 times the calculated peak containment pressure ($P_a$). For those valves sealed by IVSW, the possibility of leakage from the Containment or Reactor Coolant System to the atmosphere outside containment is eliminated because leakage will be from the IVSW system into the Containment.

The containment is designed with an allowable leakage rate not to exceed 0.1% of the containment air weight per day. The maximum allowable leakage rate is used to evaluate offsite doses resulting from a DBA. Confirmation that the leakage rate is within limit is demonstrated by the performance of a Type A leakage rate test in accordance with the Containment Leakage Rate Testing Program as required by LCO 3.6.1, "Containment." During the performance of the Type A test, no credit is taken for the IVSW System in meeting the containment leakage rate criteria. As such, in the event of a DBA without an OPERABLE IVSW System, both the whole body and thyroid offsite doses would be within the guidelines specified in 10 CFR Part 100.

Although IVSW is not needed to maintain plant releases such that the whole body and thyroid offsite doses would be within the guidelines specified in 10 CFR Part 100 based on Type A leakage testing, Indian Point 3 elected to consider IVSW as a seal system

BASES

BACKGROUND (continued)

as described in 10 CFR 50, Appendix J, (Ref. 3). This election allows leakage through CIVs sealed by IVSW to be excluded when calculating Type B and C testing results. Therefore, operation of IVSW is an implicit assumption in the calculation of post accident offsite radiation doses.

To satisfy the requirements of 10 CFR 50, Appendix J, for excluding leakage from CIVs sealed by IVSW from Type B and C limits, Technical Specifications must ensure the IVSW sealing function (i.e., both sealing water supply and nitrogen gas supply) is maintained at a pressure of 1.10 $P_a$ for at least 30 days.

Sealing water design capacity is sufficient to maintain a source of seal water at the required pressure for a minimum of 24 hours without operator intervention assuming worst case leakage and the single failure of a CIV sealed by IVSW. The requirements for a 24 hour supply of seal water under worst case conditions is satisfied by maintaining a minimum of 144 gallons in the 176 gallon capacity seal water tank.

Nitrogen gas for IVSW seal water pressurization is satisfied by having three compressed nitrogen bottles in the IVSW supply bank aligned to the IVSW supply tank.

To satisfy the requirement of 10 CFR 50, Appendix J, (Ref. 3) for maintaining the IVSW sealing function for at least 30 days, manual operator action may be required to replenish the IVSW seal water supply and/or compressed gas supply. Two sources of makeup water and two alternate sources of compressed gas with sufficient capacity to maintain the IVSW sealing function for 30 days are available. The two sources of makeup water are the primary water storage tank and the city water system. The two alternate sources of compressed gas are the normally isolated nitrogen gas bottles in the nitrogen supply bank and the ability to refill the IVSW nitrogen supply bottles from the plant Nitrogen System. Manual operations required to supply makeup water and gas to the IVSW system are performed in an area that is accessible during an accident.

BASES

BACKGROUND (continued)

The IVSW tank is instrumented to provide local indication of pressure and water level. Low water level, low pressure and high pressure in the IVSW supply tank are alarmed.

The IVSW System distribution piping consists of five headers. Three of the five IVSW headers are pressurized by opening either of a pair of normally closed air operated header injection valves. These valves open automatically on a containment Phase "A" isolation signal to admit seal water to the associated CIVs. One of the five IVSW headers is pressurized by opening either of a pair of normally closed, air-motor operated, header injection valves. These valves open automatically on a containment Phase "A" isolation signal to admit seal water to the associated CIVs. One IVSW header is used to supply seal water to CIVs on process lines that are not automatically closed on a containment Phase "A" isolation signal. This header is normally pressurized by the IVSW System with a normally closed manual or air-motor operated isolation valve for each pair of CIVs served by this IVSW header.

Redundant automatic header injection valves in parallel ensures the IVSW header is pressurized if there is a failure of one injection valve. Each of the two automatic header injection valves in each pair are actuated from separate and independent signals.

A related system, the Isolation Valve Seal Gas System, is not credited as a seal system as described in 10 CFR 50, Appendix J, and is not addressed by this Technical Specification. This system uses the nitrogen bank used to supply the IVSW System to supply high pressure nitrogen that may be used to seal lines subjected to pressure in excess of the 150 psig IVSW design pressure due to operation of the recirculation pumps. This system is manually initiated during the post accident recovery phase and is not part of the IVSW System.

BASES

APPLICABLE SAFETY ANALYSES

The IVSW System LCO was derived from the requirement related to the control of leakage from the containment during major accidents. This LCO is intended to ensure the actual containment leakage rate is maintained within the maximum value assumed in the safety analyses. As part of the containment boundary, containment isolation valves function to support the leak tightness of the containment. The IVSW System assures the effectiveness of certain containment isolation valves by providing a water seal pressurized to ≥ 1.1 times the maximum peak containment accident pressure at the valves and thereby reducing containment leakage. As such, the IVSW System is considered a seal system as described in 10 CFR 50, Appendix J. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBA that results in a release of radioactive material within containment is a loss of coolant accident (LOCA)(Ref. 2). The DBAs assume that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, $L_a$. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power) and containment isolation valve stroke time. The IVSW System actuates on a containment isolation signal and functions within 60 seconds to help reduce containment leakage within the allowable design leakage rate value, $L_a$.

The Isolation Valve Seal Water System satisfies Criterion 3 of 10 CFR 50.36.

LCO

OPERABILITY of the IVSW System is based on the system's capability to supply seal water to selective containment isolation valves within the time assumed in the applicable safety analyses and to ensure pressure is maintained for at least 30 days. This requires the IVSW tank be maintained with an adequate volume of water, an air or nitrogen overpressure sufficient to provide the motive force to move the water to the applicable

BASES

LCO (continued)

penetration, piping to provide an OPERABLE flow path and two air operated header injection valves on each automatically actuated branch header.

APPLICABILITY

The IVSW System is required to be OPERABLE in MODES 1, 2, 3, and 4 because a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the IVSW System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

With one IVSW System header inoperable, a portion of the CIVs serviced by IVSW may not receive seal water at the required pressure and volume for effective sealing. However, the CIVs are OPERABLE and will still close, the affected CIVs provide adequate isolation to meet containment isolation requirements without IVSW during the most recent Type A test, and the number of CIVs affected by the failure of one IVSW header is small compared to the total number of CIVs. Therefore, the 7 days is allowed to restore the IVSW System header to OPERABLE status.

With one IVSW automatic actuation valve inoperable, the IVSW function is still available because the redundant automatic actuation valve is OPERABLE. Therefore, the 7 days is allowed to restore the IVSW automatic actuation valve to OPERABLE status.

B.1

With the IVSW system inoperable for reasons other than Condition A, the effectiveness of CIVs sealed by IVSW may be compromised. This Condition may result from failure to meet any of the the surveillance requirements needed to verify Operability of IVSW or the inoperability of multiple IVSW headers or automatic actuation

BASES

ACTIONS

B.1 (continued)

devices. However, the CIVs are OPERABLE and will still close and the affected CIVs provide adequate isolation to meet containment isolation requirements without IVSW during the most recent Type A test. Additionally, except in the unusual case where inoperability is the result of failure to meet SR 3.6.9.5, the affected CIVs have demonstrated the ability to satisfy IVSW leakage requirements using IVSW seal water in lieu of meeting Type C testing requirements. Therefore, the 24 hours is allowed to restore the IVSW System to OPERABLE status.

C.1 and C.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

This SR verifies the IVSW tank has the necessary pressure to provide motive force to the seal water. A 47 psig pressure is sufficient to ensure the containment penetration flowpaths that are sealed by the IVSW System are maintained at a pressure equal to or greater than 1.1 times the calculated peak containment internal pressure (Pa) related to the design bases accident. Verification of the IVSW tank pressure on a Frequency of once per 24 hours is acceptable because operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.2

This SR ensures the capability of the IVSW nitrogen source to pressurize the IVSW system as needed to support IVSW operation for a minimum of 30 days. Verification of the IVSW tank pressure on a Frequency of once per 24 hours is acceptable because operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.6.9.3

This SR verifies the IVSW tank has an initial volume of water necessary to provide seal water to the containment isolation valves served by the IVSW System for a period of at least 24 hours assuming the failure of one CIV and the maximum allowed leakage past other CIVs served by IVSW. Verification of IVSW tank level on a Frequency of once per 24 hours is acceptable since tank level is monitored by installed instrumentation and will alarm in the Primary Auxilliary Building prior to level decreasing to 80 gallons.

SR 3.6.9.4

This SR verifies the stroke time of each automatic IVSW header injection solenoid valve is within limits. The frequency is specified by the Inservice Testing Program, and previous operating experience has shown that these valves usually pass the required test when performed.

SR 3.6.9.5

This SR ensures that automatic header injection valves actuate to the correct position on a simulated or actual signal. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.6

Integrity of the IVSW seal boundary is important in providing assurance that the design leakage value required for the system to perform its sealing function is not exceeded. This testing is performed in accordance with the requirements, Frequency and acceptance criteria established in Specification 5.5.15, Containment Leakage Rate Testing Program. This program was established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by IP3 specific approved exemptions. This program conforms to guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995."

REFERENCES

1. FSAR, Section 6.
2. FSAR, Chapter 14.
3. 10 CFR 50, Appendix J.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.9:
# "Isolation Valve Seal Water (IVSW) System"

## PART 2:

## CURRENT TECHNICAL SPECIFICATION PAGES

## Annotated to show differences between CTS and ITS

**CTS pages and associated TSCRs annotated for this ITS Specification :**

| CTS Page No. | Effective Amendment | Annotated Amendment | TSCR No. | TSCR Description | ITS Status of TSCR |
|---|---|---|---|---|---|
| 3.3-7 | 132 | 132 | No TSCRs | No TSCRs for this Page | N/A |
| 4.4-4 | 174 | 174 | No TSCRs | No TSCRs for this Page | N/A |
| 4.4-9 | 174 | 174 | No TSCRs | No TSCRs for this Page | N/A |

ITS 3.6.9

A.1

C. Isolation Valve Seal Water System (IVSWS) Mode 1, 2, 3 and 4 — A.3

1. The ~~reactor shall not be brought above cold shutdown~~ ~~unless~~ the following requirements are met:

LCO 3.6.9

a. The IVSWS shall be operable.

SR 3.6.9.1
SR 3.6.9.3

b. The IVSW tank shall be maintained and verified every 7 days — M.2 at a minimum pressure of 47 psig and contain a minimum of 144 gallons of water.

Cond A & Req. Act A.1

2. The requirements of 3.3.C.1 may be modified to allow any one of the following components to be inoperable at any one time:

a. Any one header of the IVSWS may be inoperable for a period not to exceed 7 consecutive days.

b. Any valve required for the functioning of the system during and following accident conditions provided it is restored to an operable status within 7 days and all valves in the system that provide a duplicate function are operable.

Cond. C

3. If the IVSW System is not restored to an operable status within the time period specified, then:

Req. Act C.1

a. If the reactor is critical, it shall be brought to 6 hours — A.4 the hot shutdown condition ~~utilizing normal operating procedures~~. ~~The shutdown shall start no later than at the end of the specified time period~~ — A.4

~~b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.~~

~~c. In either case, if the IVSW System is not restored to an operable status within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.~~

Add Req. Act. C.2 — M.1

Add Cond B and associated Req. Act — L.1

SEE ITS 5.15, Containment Leak Rate Test Program

E. Containment Isolation Valves

| | | |
|---|---|---|
| SEE ITS 3.6.3 | 1. | Verify the combined leakage rate for all containment bypass leakage paths, Table 4.4-1 lists required isolation valves, is ≤ 0.6La when pressurized ≥ Pa, in accordance with the Containment Leakage Rate Testing Program. |
| SR 3.6.9.5 | ~~2.~~ | Verify the leakage rate of water from the Isolation Valve Seal Water System is ≤ 14,700 cc/hr when pressurized ≥ 1.1 Pa, in accordance with the Containment Leakage Rate Testing Program. |
| SEE CTS RELOCATED | 3. | Verify the leakage rate of water into the containment from isolation valves sealed with the service water system is ≤ 0.36 gpm per fan cooler unit when pressurized ≥ 1.1 Pa, in accordance with the Containment Leakage Rate Testing Program. |

Add SR 3.6.9.2
Add SR 3.6.9.4
Add SR 3.6.9.5

M.3

A.1

~~The containment isolation valve surveillance requirement ensures that the combined leakage rate of all containment bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves, and, when pressurizing between valves, the total leakage of all the valves being tested) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. The Frequency is required by the Containment Leakage Rate Testing Program. This surveillance requirement simply imposes additional acceptance criteria. The service water lines to the containment fan cooler units and the lines supplied water by the Isolation Valve Seal Water System (IVSWS)[6] have containment isolation valves that are hydrostatically tested. Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of offsite doses are met. The Frequency is required by the Containment Leakage Rate Testing Program. Sufficient water is available in the Isolation Valve Seal Water System, Primary Water System, Service Water System, Residual Heat Removal System, and the City Water System to assure a sealing function for at least 30 days. The leakage limit for the Isolation Valve Seal Water System is consistent with the design capacity of the Isolation Valve Seal Water supply tank. The seal water provided by these systems is credited with limiting containment leakage (the measured leakage is not considered part of the allowable containment leakage).~~

~~The 350 psig test pressure, achieved either by normal Residual Heat Removal System operation or hydrostatic testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the hydrostatic test pressure for the containment sump return line of 100 psig gives an adequate margin over the highest pressure within the line after a design basis accident. A recirculation system leakage of 2 gal./hr. will limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident.~~

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

# Technical Specification 3.6.9:
# "Isolation Valve Seal Water (IVSW) System"

## PART 3:

## DISCUSSION OF CHANGES

## Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.6.9 - Isolation Valve Seal Water System (IVSWS)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.3.C.1 establishes the Applicability for the Isolation Valve Seal Water (IVSW) System as whenever the reactor is above cold shutdown. ITS 3.6.9 maintains this Applicability by requiring that IVSW is Operable in Modes 1, 2, 3 and 4 (i.e. whenever the reactor is above cold shutdown). This is an administrative change with no impact on safety because there is no change to the CTS Applicability.

A.4 CTS 3.3.C.3.a specifies that, if the IVSW System is not restored to Operable within a specified completion time, then the reactor shall be brought to hot shutdown (i.e., Mode 3) within an additional 48 hours utilizing normal operating procedures and that the shutdown shall start no later than the end of the specified period (i.e., Completion Time to restore to Operable). Under the same conditions, ITS LCO 3.6.9, Required Action C.1, require the plant be in Mode 3 in 6 hours. This is an administrative change with no impact on safety because the Completion Time of 6 hours is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. Additionally, as explained in ITS 1.3, Completion Times, completion time clocks always start as soon as applicable requirements are not met.

MORE RESTRICTIVE

M.1 CTS 3.3.C.3.a specifies that if the IVSW System is not restored to an operable status within the time period specified, then the reactor shall be brought to the hot shutdown condition utilizing normal operating procedures (See ITS 3.6.9, DOC A.4). Thereafter, CTS 3.3.C.3.b and c allow an additional 48 hours to restore IVSW before the reactor must be placed in cold shutdown (i.e., Mode 5). ITS LCO 3.6.9, Required Action C.1, maintains the requirement to be in hot shutdown condition utilizing normal operating procedures (See ITS 3.6.9, DOC A.4); however, Required Action C.2, requires that the plant be in Mode 5 in 36 hours (i.e., the 48 hour allowable out of service time in Mode 3 is deleted. This change is needed and is acceptable because the plant must be placed outside the applicable Mode promptly when LCO requirements are not met. This change has no adverse impact on safety.

M.2 CTS 3.3.C.1.b requires that IVSW tank be maintained at a minimum pressure of 47 psig and contain a minimum of 144 gallons of water; however, there is no requirement for periodic verification that these requirements are met. ITS LCO 3.6.9 maintains the requirement to keep these parameters within specified limits; however, ITS SR 3.6.9.1 and SR 3.6.9.3 are added to require verification every 7 days that these parameters are within limits. Verification of the IVSW tank pressure and level on a Frequency of once per 7 days is acceptable because tank level and pressure are monitored by installed instrumentation and will alarm in the Primary Auxiliary Building when below required limits]. This more restrictive change is acceptable because it requires periodic verification that IVSW system parameters are within limits with no impact on any other aspect of plant safety.

M.3 CTS 3.3.C does not include a specific requirement for periodic verification that IVSW nitrogen supply is available to support IVSW operation. ITS SR 3.6.9.2 adds a requirement to verify every 7 days that a nitrogen bottle with a minimum of 150 psig is available to support operation of the IVSW system.

CTS 3.3.C does not include a specific requirement to verify the opening time of automatic valves in the IVSW System actuates. ITS SR 3.6.9.4 adds a requirement to verify valves actuate within the required time limits as required by the Inservice Testing Program.

CTS 3.3.C does not include a specific requirement to verify each automatic valve in the IVSW System actuates to the correct position on an actual or simulated actuation signal. ITS SR 3.6.9.5 adds a requirement to verify proper operation of each automatic valve in the IVSW System every 24 months.

These more restrictive changes are acceptable because they require periodic verification that IVSW system will function as required following containment isolation with no impact on any other aspect of plant safety.

LESS RESTRICTIVE

L.1 CTS 3.3.C.2 provides an allowable out of service time (AOT) if one IVSW system header inoperable or if one IVSW automatic actuation valve inoperable; however, no AOT is provided for IVSW inoperability for any other reason (e.g., water level low, nitrogen pressure low, etc). Therefore, CTS 3.0 would require immediate shutdown under these conditions. ITS LCO 3.6.9 maintains the 7 day AOT if one IVSW system header inoperable or if one IVSW automatic actuation valve inoperable; however, ITS LCO 3.6.9, Condition B and associated Required Action, establish a 24 hour AOT if IVSW is inoperable for any reason other than Condition A. This change is needed because it eliminates a requirement for an immediate plant shutdown for numerous conditions that can either be resolved very quickly or do not merit plant shutdown. This change is acceptable because the CIVs associated with IVSW are still Operable and will still close and the affected CIVs provided adequate isolation to meet containment isolation requirements without IVSW during the most recent Type A test. Additionally, except in the unusual case where inoperability is the result of failure to meet SR 3.6.9.5, the affected CIVs have demonstrated the ability to satisfy IVSW leakage requirements using IVSW seal water in lieu of meeting Type C testing requirements. Therefore, the 24 hours is allowed to restore the IVSW System to OPERABLE status. Finally, the Safety Function Determination Program (SFDP) required by ITS 5.5.14 will ensure that appropriate actions are taken if loss of IVSW results in a loss of function for one or more CIVs. This program ensures loss of safety function is detected and appropriate actions taken. Therefore, this change has no adverse impact on safety.

REMOVED DETAIL

None

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

## Technical Specification 3.6.9:
## "Isolation Valve Seal Water (IVSW) System"

## PART 4:

## No Significant Hazards Considerations
## for
## Changes between CTS and ITS
## that are
## Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

NO SIGNIFICANT HAZARDS EVALUATION
ITS SECTION 3.6.9 - Isolation Valve Seal Water System (IVSWS)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   CTS 3.3.C.2 provides an allowable out of service time (AOT) if one IVSW system header inoperable or if one IVSW automatic actuation valve inoperable; however, no AOT is provided for IVSW inoperability for any other reason (e.g., water level low, nitrogen pressure low, etc). Therefore, CTS 3.0 would require immediate shutdown under these conditions. ITS LCO 3.6.9 maintains the 7 day AOT if one IVSW system header inoperable or if one IVSW automatic actuation valve inoperable; however, ITS LCO 3.6.9, Condition B and associated Required Action, establish a 24 hour AOT if IVSW is inoperable for any reason other than Condition A. This change is needed because it eliminates a requirement for an immediate plant shutdown for numerous conditions that can either be resolved very quickly or do not merit plant shutdown.

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because the CIVs associated with IVSW are still Operable and will still close and the affected CIVs provided adequate isolation to meet containment isolation requirements without IVSW during the most recent Type A test. Additionally, except in the unusual case where inoperability is the result of failure to meet SR 3.6.9.5, the affected CIVs have demonstrated the ability to satisfy IVSW leakage requirements using IVSW seal water in lieu of meeting Type C testing requirements. Finally, the Safety Function Determination Program (SFDP) required by ITS 5.5.14 will ensure that appropriate actions are taken if loss of IVSW results in a loss of function for one or more CIVs. This program ensures loss of safety function is detected and appropriate actions taken.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in margin of safety because the CIVs associated with IVSW are still Operable and will still close and the affected CIVs provided adequate isolation to meet containment isolation requirements without IVSW during the most recent Type A test. Additionally, except in the unusual case where inoperability is the result of failure to meet SR 3.6.9.5, the affected CIVs have demonstrated the ability to satisfy IVSW leakage requirements using IVSW seal water in lieu of meeting Type C testing requirements. Finally, the Safety Function Determination Program (SFDP) required by ITS 5.5.14 will ensure that appropriate actions are taken if loss of IVSW results in a loss of function for one or more CIVs. This program ensures loss of safety function is detected and appropriate actions taken.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.6.9:
# "Isolation Valve Seal Water (IVSW) System"

## PART 5:

## NUREG-1431
## Annotated to show differences between NUREG-1431 and ITS

### Status of NUREG 1431 Generic Changes for ITS 3.6.9

This ITS Specification is based on NUREG-1431 Specification No. **IP3 UNIQUE 1x**
as modified by the following Generic Changes:

| OG No. | TSTF No. | Generic Change Description | NRC STATUS | IP3 STATUS | JD No. |
|---|---|---|---|---|---|
| NA | NA | NOT APPLICABLE. THIS IS AN IP3 UNIQUE SPECIFICATION. | Not Applicable | Not Applicable | N/A |

## 3.6 CONTAINMENT SYSTEMS

<CTS>

### 3.6.9 Isolation Valve Seal Water (IVSW) System

<3.3.C.1.a>

LCO 3.6.9 The IVSW System shall be OPERABLE.

<3.3.C.1>
<DOC A3>

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

| | CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|---|
| <3.3.c.2> | A. One IVSW system header inoperable.<br><br>OR<br><br>One IVSW automatic actuation valve inoperable. | A.1 Restore IVSW system to OPERABLE status. | 7 days |
| <DOC L.1> | B. IVSW system inoperable for reasons other than Condition A. | B.1 Restore IVSW System to OPERABLE Status. | 24 hours |
| <3.3.c.3.a><br><3.3.c.3.b><br><3.3.c.3.c><br><DOC M.1> | C. Required Action and associated Completion Time not met. | C.1 Be in MODE 3.<br><br>AND<br><br>C.2 Be in MODE 5. | 6 hours<br><br><br><br>36 hours |

SURVEILLANCE REQUIREMENTS

| <CTS> | SURVEILLANCE | | FREQUENCY |
|---|---|---|---|
| <3.3.C.1.b><br><DOC M.2> | SR 3.6.9.1 | Verify IVSW tank pressure is ≥ 47 psig. | 24 hours |
| <DOC M.3> | SR 3.6.9.2 | Verify IVSW nitrogen supply bank includes a minimum of 3 cylinders and that each cylinder has a pressure ≥ 150 psig. | 24 hours |
| <3.3.C.1.b><br><DOC M.2> | SR 3.6.9.3 | Verify the IVSW tank water volume is ≥ 144 gallons. | 24 hours |
| <DOC M.3> | SR 3.6.9.4 | Verify the opening time of each air operated header injection valve is within limits. | In accordance with the Inservice Testing Program |
| <DOC M.3> | SR 3.6.9.5 | Verify each automatic valve in the IVSW System actuates to the correct position on an actual or simulated actuation signal. | 24 months |
| <4.4.E.2> | SR 3.6.9.6 | Verify the leakage rate of water from the Isolation Valve Seal Water System in accordance with the Containment Leakage Rate Testing Program. | In accordance with the Containment Leakage Rate Testing Program. |

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.9 Isolation Valve Seal Water (IVSW) System

BASES

BACKGROUND

The Isolation Valve Seal Water (IVSW) System improves the effectiveness of certain containment isolation valves (CIVs) by providing a water seal to valve leakage paths. This is accomplished by injecting water between the seats and stem packing of globe and double-disk type isolation valves and into the piping between other closed containment isolation valves. IVSW sealing water is maintained in a seal water supply tank filled with water and pressurized with nitrogen. The IVSW System is actuated in conjunction with automatic initiation of containment isolation and is applied to CIVs in lines connected to the Reactor Coolant System or exposed to the containment atmosphere during an accident. The seal water is injected at a pressure of at least 47 psig which is > 1.1 times the calculated peak containment pressure ($P_a$). For those valves sealed by IVSW, the possibility of leakage from the Containment or Reactor Coolant System to the atmosphere outside containment is eliminated because leakage will be from the IVSW system into the Containment.

The containment is designed with an allowable leakage rate not to exceed 0.1% of the containment air weight per day. The maximum allowable leakage rate is used to evaluate offsite doses resulting from a DBA. Confirmation that the leakage rate is within limit is demonstrated by the performance of a Type A leakage rate test in accordance with the Containment Leakage Rate Testing Program as required by LCO 3.6.1, "Containment." During the performance of the Type A test, no credit is taken for the IVSW System in meeting the containment leakage rate criteria. As such, in the event of a DBA without an OPERABLE IVSW System, both the whole body and thyroid offsite doses would be within the guidelines specified in 10 CFR Part 100.

Although IVSW is not needed to maintain plant releases such that the whole body and thyroid offsite doses would be within the guidelines specified in 10 CFR Part 100 based on Type A leakage testing, Indian Point 3 elected to consider IVSW as a seal system

BASES

BACKGROUND (continued)

as described in 10 CFR 50, Appendix J, (Ref. 3). This election allows leakage through CIVs sealed by IVSW to be excluded when calculating Type B and C testing results. Therefore, operation of IVSW is an implicit assumption in the calculation of post accident offsite radiation doses.

To satisfy the requirements of 10 CFR 50, Appendix J, for excluding leakage from CIVs sealed by IVSW from Type B and C limits, Technical Specifications must ensure the IVSW sealing function (i.e., both sealing water supply and nitrogen gas supply) is maintained at a pressure of 1.10 $P_a$ for at least 30 days.

Sealing water design capacity is sufficient to maintain a source of seal water at the required pressure for a minimum of 24 hours without operator intervention assuming worst case leakage and the single failure of a CIV sealed by IVSW. The requirements for a 24 hour supply of seal water under worst case conditions is satisfied by maintaining a minimum of 144 gallons in the 176 gallon capacity seal water tank.

Nitrogen gas for IVSW seal water pressurization is satisfied by having three compressed nitrogen bottles in the IVSW supply bank aligned to the IVSW supply tank.

To satisfy the requirement of 10 CFR 50, Appendix J, (Ref. 3) for maintaining the IVSW sealing function for at least 30 days, manual operator action may be required to replenish the IVSW seal water supply and/or compressed gas supply. Two sources of makeup water and two alternate sources of compressed gas with sufficient capacity to maintain the IVSW sealing function for 30 days are available. The two sources of makeup water are the primary water storage tank and the city water system. The two alternate sources of compressed gas are the normally isolated nitrogen gas bottles in the nitrogen supply bank and the ability to refill the IVSW nitrogen supply bottles from the plant Nitrogen System. Manual operations required to supply makeup water and gas to the IVSW system are performed in an area that is accessible during an accident.

BASES

BACKGROUND (continued)

The IVSW tank is instrumented to provide local indication of pressure and water level. Low water level, low pressure and high pressure in the IVSW supply tank are alarmed.

The IVSW System distribution piping consists of five headers. Three of the five IVSW headers are pressurized by opening either of a pair of normally closed air operated header injection valves. These valves open automatically on a containment Phase "A" isolation signal to admit seal water to the associated CIVs. One of the five IVSW headers is pressurized by opening either of a pair of normally closed, air-motor operated, header injection valves. These valves open automatically on a containment Phase "A" isolation signal to admit seal water to the associated CIVs. One IVSW header is used to supply seal water to CIVs on process lines that are not automatically closed on a containment Phase "A" isolation signal. This header is normally pressurized by the IVSW System with a normally closed manual or air-motor operated isolation valve for each pair of CIVs served by this IVSW header.

Redundant automatic header injection valves in parallel ensures the IVSW header is pressurized if there is a failure of one injection valve. Each of the two automatic header injection valves in each pair are actuated from separate and independent signals.

A related system, the Isolation Valve Seal Gas System, is not credited as a seal system as described in 10 CFR 50, Appendix J, and is not addressed by this Technical Specification. This system uses the nitrogen bank used to supply the IVSW System to supply high pressure nitrogen that may be used to seal lines subjected to pressure in excess of the 150 psig IVSW design pressure due to operation of the recirculation pumps. This system is manually initiated during the post accident recovery phase and is not part of the IVSW System.

BASES

APPLICABLE SAFETY ANALYSES

The IVSW System LCO was derived from the requirement related to the control of leakage from the containment during major accidents. This LCO is intended to ensure the actual containment leakage rate is maintained within the maximum value assumed in the safety analyses. As part of the containment boundary, containment isolation valves function to support the leak tightness of the containment. The IVSW System assures the effectiveness of certain containment isolation valves by providing a water seal pressurized to ≥ 1.1 times the maximum peak containment accident pressure at the valves and thereby reducing containment leakage. As such, the IVSW System is considered a seal system as described in 10 CFR 50, Appendix J. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBA that results in a release of radioactive material within containment is a loss of coolant accident (LOCA)(Ref. 2). The DBAs assume that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, $L_a$. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power) and containment isolation valve stroke time. The IVSW System actuates on a containment isolation signal and functions within 60 seconds to help reduce containment leakage within the allowable design leakage rate value, $L_a$.

The Isolation Valve Seal Water System satisfies Criterion 3 of 10 CFR 50.36.

LCO

OPERABILITY of the IVSW System is based on the system's capability to supply seal water to selective containment isolation valves within the time assumed in the applicable safety analyses and to ensure pressure is maintained for at least 30 days. This requires the IVSW tank be maintained with an adequate volume of water, an air or nitrogen overpressure sufficient to provide the motive force to move the water to the applicable

BASES

LCO (continued)

penetration, piping to provide an OPERABLE flow path and two air operated header injection valves on each automatically actuated branch header.

APPLICABILITY

The IVSW System is required to be OPERABLE in MODES 1, 2, 3, and 4 because a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the IVSW System is not required to be OPERABLE in MODE 5 and 6.

ACTIONS

A.1

With one IVSW System header inoperable, a portion of the CIVs serviced by IVSW may not receive seal water at the required pressure and volume for effective sealing. However, the CIVs are OPERABLE and will still close, the affected CIVs provide adequate isolation to meet containment isolation requirements without IVSW during the most recent Type A test, and the number of CIVs affected by the failure of one IVSW header is small compared to the total number of CIVs. Therefore, the 7 days is allowed to restore the IVSW System header to OPERABLE status.

With one IVSW automatic actuation valve inoperable, the IVSW function is still available because the redundant automatic actuation valve is OPERABLE. Therefore, the 7 days is allowed to restore the IVSW automatic actuation valve to OPERABLE status.

B.1

With the IVSW system inoperable for reasons other than Condition A, the effectiveness of CIVs sealed by IVSW may be compromised. This Condition may result from failure to meet any of the surveillance requirements needed to verify Operability of IVSW or the inoperability of multiple IVSW headers or automatic actuation

BASES

ACTIONS

B.1 (continued)

devices. However, the CIVs are OPERABLE and will still close and the affected CIVs provide adequate isolation to meet containment isolation requirements without IVSW during the most recent Type A test. Additionally, except in the unusual case where inoperability is the result of failure to meet SR 3.6.9.5, the affected CIVs have demonstrated the ability to satisfy IVSW leakage requirements using IVSW seal water in lieu of meeting Type C testing requirements. Therefore, the 24 hours is allowed to restore the IVSW System to OPERABLE status.

C.1 and C.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

This SR verifies the IVSW tank has the necessary pressure to provide motive force to the seal water. A 47 psig pressure is sufficient to ensure the containment penetration flowpaths that are sealed by the IVSW System are maintained at a pressure equal to or greater than 1.1 times the calculated peak containment internal pressure (Pa) related to the design bases accident. Verification of the IVSW tank pressure on a Frequency of once per 24 hours is acceptable because operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.2

This SR ensures the capability of the IVSW nitrogen source to pressurize the IVSW system as needed to support IVSW operation for a minimum of 30 days. Verification of the IVSW tank pressure on a Frequency of once per 24 hours is acceptable because operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.6.9.3

This SR verifies the IVSW tank has an initial volume of water necessary to provide seal water to the containment isolation valves served by the IVSW System for a period of at least 24 hours assuming the failure of one CIV and the maximum allowed leakage past other CIVs served by IVSW. Verification of IVSW tank level on a Frequency of once per 24 hours is acceptable since tank level is monitored by installed instrumentation and will alarm in the Primary Auxilliary Building prior to level decreasing to 80 gallons.

SR 3.6.9.4

This SR verifies the stroke time of each automatic IVSW header injection solenoid valve is within limits. The frequency is specified by the Inservice Testing Program, and previous operating experience has shown that these valves usually pass the required test when performed.

SR 3.6.9.5

This SR ensures that automatic header injection valves actuate to the correct position on a simulated or actual signal. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.6

Integrity of the IVSW seal boundary is important in providing assurance that the design leakage value required for the system to perform its sealing function is not exceeded. This testing is performed in accordance with the requirements, Frequency and acceptance criteria established in Specification 5.5.15, Containment Leakage Rate Testing Program. This program was established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by IP3 specific approved exemptions. This program conforms to guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995."

REFERENCES

1. FSAR, Section 6.
2. FSAR, Chapter 14.
3. 10 CFR 50, Appendix J.

# Indian Point 3
# Improved Technical Specifications (ITS)
# Conversion Package

# Technical Specification 3.6.9:
# "Isolation Valve Seal Water (IVSW) System"

## PART 6:

## Justification of Differences between

## NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.6.9 - Isolation Valve Seal Water System (IVSWS)

This ITS Specification is unique to IP3. Therefore, there is no markup of NUREG-1431.