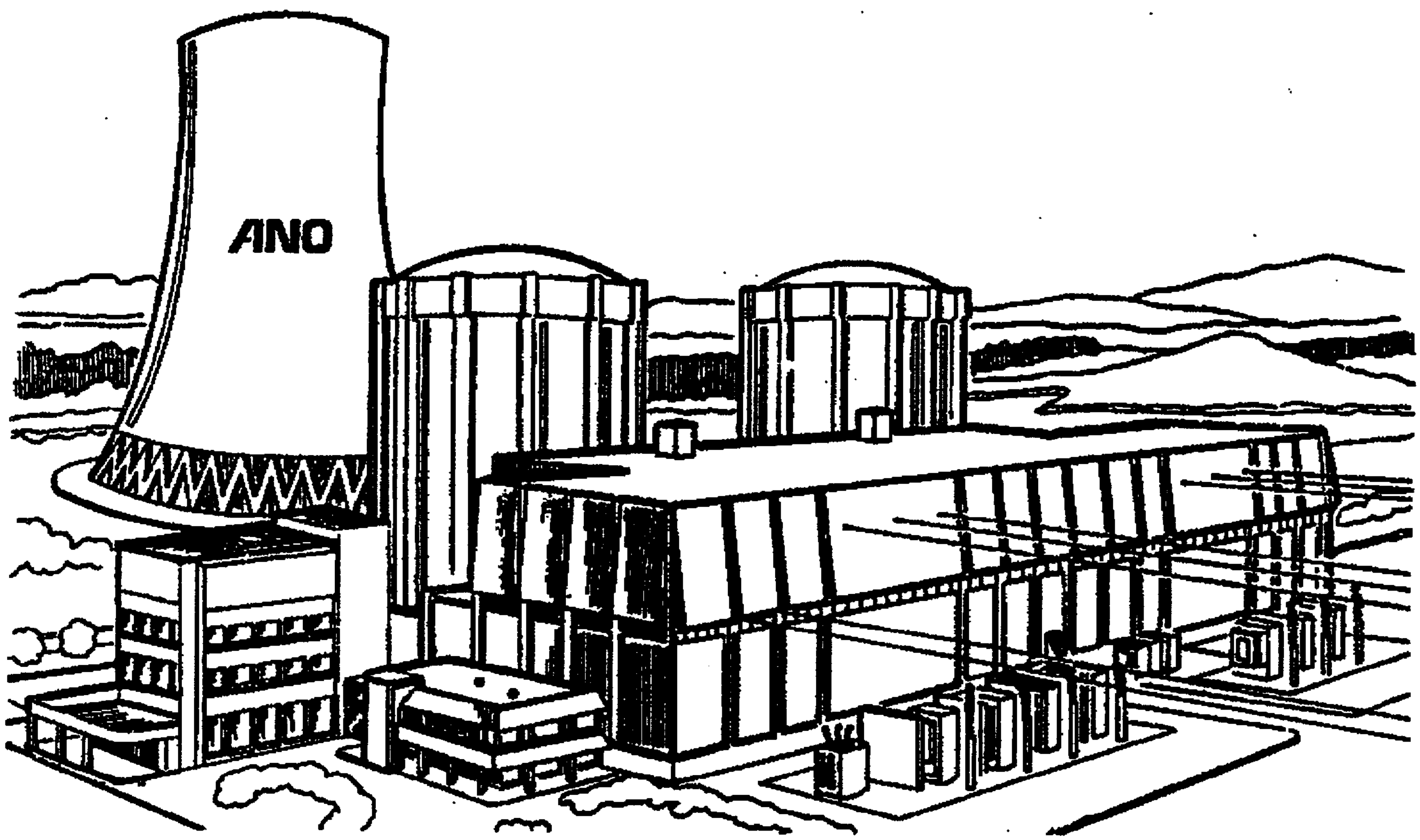
ARKANSAS NUCLEAR ONE - UNIT 1

IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL



VOLUME 7 OF 7

Entergy

January 28, 2000

5.5.2
5.5.3

Not addressed by TSIP.

(4) Physical Protection

EOI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10CFR73.55 (51 FR 27817 and 27822) and to the authority of 10CFR50.90 and 10CFR50.54(p). The plan, which contains Safeguards Information protected under 10CFR73.21, is entitled: "Arkansas Nuclear One Industrial Security Plan," with revisions submitted through August 4, 1995. The Industrial Security Plan also includes the requirements for guard training and qualification in Appendix A and the safeguards contingency events in Chapter 7. Changes made in accordance with 10CFR73.55 shall be implemented in accordance with the schedule set forth therein.

5.5.2 (5) Systems Integrity Not used.

EOI shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

LA4
SAR

5.5.3 (6) Iodine Monitoring Not used.

EOI shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration ~~in vital areas~~ under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

5.10

5.5.10 (7) Secondary Water Chemistry Monitoring Not used.

This program provides controls for A ~~secondary water chemistry monitoring program shall be implemented~~ to ~~minimize~~ inhibit steam generator tube degradation. This program shall include: A1

1. Identification of a sampling schedule for the critical ~~parameters~~ variables and control points for these ~~parameters~~ variables;

2. Identification of the procedures used to measure the values of the critical ~~parameters~~ variables;

3. Identification of process sampling points;

4. Procedures for the recording and management of data;

5. Procedures defining corrective actions for off-control point chemistry conditions; and

6. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate a corrective action.

Not addressed by TSIP.

(8) FIRE PROTECTION

EOI shall implement and maintian in effect all provisions of the approved Fire Protection Program as described in Amendment 9A to the Safety Analysis Report and as approved in the Safety Evaluation dated March 31, 1992, subject to the following provision:

The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3. This license is effective as of the date of issuance and shall expire at midnight, May 20, 2014.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by:
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:
Appendices A and B - Technical Specifications

Date of Issuance: May 21 1974

1.0 USE AND APPLICATION

1.1 Definitions

Note

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

A1

~~1. Definitions~~

~~The following terms are defined for uniform interpretation of these specifications.~~ A1

RATED THERMAL POWER (RTP)

1.1 RATED THERMAL POWER (RTP)

Rated power is a steady state reactor core output of 2568 MWt. A1

1.2 ~~REACTOR OPERATING CONDITIONS~~ A1

Table 1.1-1 MODE 5 & Note (b)

1.2.1 Cold Shutdown <Apply Table 1.1-1; Note (b)> A4

The reactor is in the cold shutdown condition when it is subcritical by at least 1 percent Δk/k and T_{avg} is no more than 200 F. ~~Pressure is defined by Specification 3.1.2.~~ A5

Table 1.1-1 MODE 4 & Note (b)

1.2.2 Hot Shutdown <Apply Table 1.1-1; Note (b)> A4

The reactor is in the hot shutdown condition when it is subcritical by at least 1 percent Δk/k and T_{avg} is ~~at or~~ greater than ~~525 F.~~ 200°F. less than 280°F and A6

~~1.2.3 Reactor Critical~~

~~The reactor is critical when the neutron chain reaction is self-sustaining and K_{eff} = 1.0.~~ A10

Table 1.1-1 MODE 3

1.2.4 Hot Standby A4

The reactor is in the hot standby condition when all of the following conditions exist:

A. T_{avg} is greater than ~~525 F.~~ or equal to 280°F. A7

B. The ~~reactor is critical.~~ reactivity condition is <0.99.

~~C. Indicated neutron power on the power range channels is less than 2 percent of rated power.~~

Table 1.1-1 MODE 1 & Note (a)

1.2.5 Power Operation <Apply Table 1.1-1; Note (a)> A4

The reactor is in a power operating condition when the indicated neutron power is above ~~2~~ 5 percent of rated power ~~as indicated on the power range channels.~~ A12 A13

Table 1.1-1 MODE 6 & Note (c)

1.2.6 Refueling ~~Shutdown~~ <Apply Table 1.1-1; Note (c)> A4

The reactor is in the refueling ~~shutdown~~ condition when ~~, even with all rods removed, the reactor would be subcritical by at least 1 percent Δk/k and the coolant temperature at the decay heat removal pump suction is at the~~ one or more reactor vessel head closure bolts is/are less than fully tensioned. A14

~~refueling temperature (normally 140F).~~ ~~Pressure is defined by Specification 3.1.2.~~ ~~A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.~~ A14 A5 A1

CORE ALTERATION

~~1.2.7 Refueling Operation~~ <Add CORE ALTERATION> A1

~~An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.~~

Table 1.1-1 MODE 2 & Note (a)

1.2.8 Startup <Apply Table 1.1-1, Note (a)> A4

The reactor shall be considered in the startup mode when the ~~shutdown margin is reduced with the intent of going critical.~~ reactivity condition is ≥ 0.99 and the THERMAL POWER is ≤ 5% RTP. A13

OPERABLE-OPERABILITY

1.3 OPERABLE - OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) ~~.~~ , and when ~~Implicit in this definition shall be the assumption that~~ all necessary attendant instrumentation, controls, normal ~~and~~ or emergency electrical power ~~sources~~, cooling and ~~or~~ seal water, lubrication and ~~or~~ other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified function~~s~~(s) are also capable of performing their related support function(s). A1 A8 A1

~~1.4 PROTECTION INSTRUMENTATION LOGIC~~ A1

~~1.4.1 Instrument Channel~~

~~An instrument channel is the combination of sensor, wires, amplifiers and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control and/or protection. An instrument channel may be either analog or digital.~~ A10

~~1.4.2 Reactor Protection System~~

~~The reactor protection system is shown in Figures 7-1 and 7-9 of the FSAR. It is that combination of protective channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protective trip breakers and activating relays or coils.~~ A10

~~A protection channel, as shown in Figure 7-1 of the FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply~~

<Add ACTIONS definition> A11

<Add MODE definition> A11

1.1

< Add LEAKAGE DEFINITION > A11

~~units, amplifiers and bistable modules provided for every reactor protection safety parameter, is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. Each protection channel includes two key operated bypass switches, a protection channel bypass switch and a shutdown bypass switch.~~ A10

~~1.4.4 Reactor Protection System Logic~~

~~This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as shown in Figure 7-1 of the FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels.~~ A10

~~1.4.5 Safety Features System~~

~~This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7-6 of the FSAR. The digital sub-system is wired to provide appropriate signals for the actuation of redundant safety features equipment on a two-of-three basis for any given parameter.~~ A10

~~1.4.6 Degree of Redundancy~~

~~The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.~~ A10

~~1.5 INSTRUMENTATION SURVEILLANCE~~ A1

1.5.1 Trip Test

A trip test is a test of logic elements in a protection channel to verify their associated trip action. A9

1.5.2 Channel ~~Test~~ FUNCTIONAL Test < CHANNEL FUNCTIONAL TEST DEFINITION AS PRESENTED IN THE ITS. > A9

CHANNEL FUNCTIONAL TEST

A channel test is the injection of an internal or external test signal into the channel to verify its proper response, including alarm and/or trip initiating action, where applicable.

1.5.3 ~~Instrument~~ Channel Check <CHANNEL CHECK DEFINITION>

CHANNEL CHECK

An instrument channel check is a verification of acceptable instrument performance by observation of its behavior and/or state; this verification includes comparison of output and/or state of independent channels measuring the same variable. A1

< Add CONTROL RODS DEFINITION > A11

< Add AXIAL POWER SHAPING RODS DEFINITION > A11

1.1

<Add PHYSICS TESTS definition> A11

<Add CHANNEL CALIBRATION definition>

1.5.4 ~~Instrument~~ Channel Calibration

~~An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip and shall be deemed to include the channel test.~~ A1

1.5.5 ~~Heat Balance Check~~

~~A heat balance check is a comparison of the indicated neutron power and core thermal power.~~ A10

1.5.6 ~~Heat Balance Calibration~~

~~An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a weighted primary and secondary heat balance considering all heat losses.~~ A16 Between 0 and 15% power, only the primary heat balance is considered. From 15 to 100% power the heat balance is weighted linearly with only the secondary heat balance being considered at 100% power.

<LATER> (3.3A) LATER

~~1.6 POWER DISTRIBUTION~~ A1

QUADRANT POWER TILT (QPT)

1.6.1 Quadrant Power Tilt CARS (QPT)

Quadrant power tilt (QPT) shall be defined by the following equation and is expressed as a percentage A1

$$QPT = 100\left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1\right)$$

AXIAL POWER IMBALANCE

1.6.2 ~~Reactor~~ AXIAL Power Imbalance CARS

~~Reactor~~ AXIAL power imbalance ~~is~~ shall be the power in the top half of the core minus the power in the bottom half of the core, expressed as a percentage of RATED THERMAL POWER (RTP) ~~rated power~~. ~~Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.~~ A1

<Add THERMAL POWER definition> A11

<Add ALLOWABLE THERMAL POWER definition>

<LATER>
(1.0)

1.5.4 **Instrument Channel Calibration**

An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip and shall be deemed to include the channel test.

1.5.5 **Heat Balance Check**

A heat balance check is a comparison of the indicated neutron power and core thermal power.

1.5.6 **Heat Balance Calibration**

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a weighted primary and secondary heat balance considering all heat losses. Between 0 and 15% power, only the primary heat balance is considered. From 15 to 100% power the heat balance is weighted linearly with only the secondary heat balance being considered at 100% power.

LATER

LA1
Bases

<LATER>
(1.0)

1.6 POWER DISTRIBUTION

1.6.1 **Quadrant Power Tilt**

Quadrant power tilt shall be defined by the following equation and is expressed as a percentage

$$100\left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1\right)$$

1.6.2 **Reactor Power Imbalance**

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

LATER

1.1

<LATER>
(3.6)

1.7 ~~REACTOR BUILDING~~

~~Reactor building integrity exists when the following conditions are satisfied:~~

~~a. The equipment hatch is closed and sealed and both doors of the personnel lock and emergency lock are closed and sealed, or b. below.~~

~~b. At least one door on each of the personnel lock and emergency lock is closed and sealed during personnel access or repair.~~

~~c. All non-automatic reactor building isolation valves and blind flanges are closed as required.~~

~~d. All automatic reactor building isolation valves are operable or deactivated in the closed position.~~

~~e. The reactor building leakage determined at the last testing interval satisfies Specification 4.4.1.~~

LATER

1.8 ~~FIRE SUPPRESSION WATER SYSTEM~~

~~The fire suppression water system consists of: water sources, pumps, and distribution piping with associated sectionalizing isolation valves. Such valves include the hose standpipe shutoff valves and the first valve ahead of the water flow alarm device on each sprinkler system.~~

A10

STAGGERED TEST BASIS

1.9 STAGGERED TEST BASIS

A staggered test basis shall consist ~~of~~:

~~a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals.~~

~~b. The testing of one system, subsystem, train or designated component at the beginning of each subinterval.~~

A15

of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.

3.6.1
3.6.2
3.6.3

1.7 REACTOR BUILDING

Reactor building integrity exists when the following conditions are satisfied: A1 A18

is OPERABLE

3.6.1 LCO
3.6.2 LCO

a. The equipment hatch ~~is closed and sealed and both doors of~~ the personnel lock and emergency lock ~~are closed and sealed, or b. below.~~ are OPERABLE LA1 Bases A18 LA1 Bases

3.6.2 LCO

b. ~~At least one door on each of the~~ personnel lock and emergency lock ~~is closed and sealed~~ during personnel access or repair. are OPERABLE A18

3.6.3 LCO
3.6.3 ACT. Note 1

c. All non-automatic reactor building isolation valves ~~and blind flanges~~ ~~are closed as required.~~ manual valves, check valves A19 are OPERABLE A18 LA1 Bases

3.6.3 LCO

d. All automatic reactor building isolation valves are operable or deactivated in the closed position.

SR 3.6.1.1

e. The reactor building leakage ~~determined at the last testing interval satisfies Specification 4.4.1.~~ is in accordance with the RBLRTP A20 LA1 Bases

<Later> (1.0)

~~1.8 FIRE SUPPRESSION WATER SYSTEM~~

~~The fire suppression water system consists of: water sources, pumps, and distribution piping with associated sectionalizing isolation valves. Such valves include the hose standpipe shutoff valves and the first valve ahead of the water flow alarm device or each sprinkler system.~~

~~1.9 STAGGERED TEST BASIS~~

~~A staggered test basis shall consist of:~~

~~a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals.~~

~~b. The testing of one system, subsystem, train or designated component at the beginning of each subinterval.~~

Later

1.10 Dose Equivalent I-131

Dose Equivalent I-131

The Dose Equivalent I-131 shall be the concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." A1

1.11 ~~Liquid Radwaste Treatment System~~

~~A Liquid Radwaste Treatment System is a system designed and used for holdup, filtration, and/or demineralization of radioactive liquid effluents prior to their release to the evironment.~~

1.12 ~~Purge - Purging~~

~~Purge or Purging is the controlled process of discharging air or gas from a confinement to reduce the airborne radioactivity concentration in such a manner that replacement air or gas is required to purify the confinement.~~

1.13 ~~Member(s) of the Public~~

~~Member(s) of the Public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.~~ A10

1.14 ~~Exclusion Area~~

~~The exclusion area is that area surrounding ANO within a minimum radius of .65 miles of the reactor buildings and controlled to the extent necessary by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.~~

1.15 ~~Unrestricted Area~~

~~An unrestricted area shall be any area beyond the exclusion area boundary.~~

1.16 Core Operating Limits Report

COLR

The CORE OPERATING LIMITS REPORT is the ANO-1 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Technical Specification ~~6.12.3~~ 5.6.5. Plant operation within these operating limits is addressed in individual specifications. A1

<Add SHUTDOWN MARGIN Definition> A11

2.0

2\. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

2.1 APPL Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow when the reactor is critical. A1 M1

MODES 1 and 2

Objective

To maintain the integrity of the fuel cladding. A1

Specification

2.1.1.1 2.1.1 The maximum local fuel pin centerline temperature shall be $\leq 5080 - (6.5 \times 10^{-3} \times (\text{Burnup, MWD/MTU})°F)$ for TACO2 applications and $\leq 4642 - (5.8 \times 10^{-3} \times (\text{Burnup, MWD/MTU})°F)$ for TACO3 applications. Operation within this limit is ensured by compliance with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR. edit LA1 Bases

2.1.1.2 2.1.2 The departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation. Operation within this limit is ensured by compliance with Specification 2.1.3 and with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR. LA1 Bases

2.1.1.3 2.1.3 Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). A2

3.4.1

<Add 3.4.1> M10

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

<LATER> (2.0)

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow when the reactor is critical. LATER

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The maximum local fuel pin centerline temperature shall be $\leq 5080 - (6.5 \times 10^{-3} \times$ (Burnup, MWD/MTU)°F) for TACO2 applications and $\leq 4642 - (5.8 \times 10^{-3} \times$ (Burnup, MWD/MTU)°F for TACO3 applications. Operation within this limit is ensured by compliance with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.2 The departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation. Operation within this limit is ensured by compliance with Specification 2.1.3 and with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.3 Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR.

A2

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

2.0

(A2)

~~A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure for the allowable RC pump combination has been considered in determining the Variable Low RCS Pressure-Temperature Protective Limits.~~

~~The Variable Low RCS Pressure-Temperature Protective Limits presented in the COLR represent the conditions at which the DNBR is greater than or equal to the minimum allowable DNBR for the limiting combination of thermal power and number of operating reactor coolant pumps which is based on the nuclear power peaking factors (3) as specified in the COLR with potential fuel densification effects.~~

~~The Axial Power Imbalance Protective Limits in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:~~

1. ~~The DNBR limit produced by the limiting combination of the radial peak, axial peak, and position of the axial peak.~~

2. ~~The combination of radial and axial peak that prevents central fuel melting at the hot spot as given in the COLR.~~

~~Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.~~

~~The flow rates for the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop.~~

~~The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive of all possible reactor coolant pump maximum thermal power combinations as specified in the COLR. The Variable Low RCS Pressure-Temperature Protective Limits in the COLR represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. If the actual pressure/temperature point is below and to the right of the pressure/temperature line, the Variable Low RCS Pressure-Temperature Protective Limit is exceeded. The local quality at the point of minimum DNBR is less than 22 percent (BAW-2)(1) or 26 percent (BWC)(2).~~

3.4.1

A2

A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure for the allowable RC pump combination has been considered in determining the Variable Low RCS Pressure-Temperature Protective Limits.

The Variable Low RCS Pressure-Temperature Protective Limits presented in the COLR represent the conditions at which the DNBR is greater than or equal to the minimum allowable DNBR for the limiting combination of thermal power and number of operating reactor coolant pumps which is based on the nuclear power peaking factors (3) as specified in the COLR with potential fuel densification effects.

The Axial Power Imbalance Protective Limits in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The DNBR limit produced by the limiting combination of the radial peak, axial peak, and position of the axial peak.

2. The combination of radial and axial peak that prevents central fuel melting at the hot spot as given in the COLR.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The flow rates for the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop.

The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive of all possible reactor coolant pump maximum thermal power combinations as specified in the COLR. The Variable Low RCS Pressure-Temperature Protective Limits in the COLR represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. If the actual pressure/temperature point is below and to the right of the pressure/temperature line, the Variable Low RCS Pressure-Temperature Protective Limit is exceeded. The local quality at the point of minimum DNBR is less than 22 percent (BAW-2)(1) or 26 percent (BWC)(2).

2.0

Using a local quality limit of 22 percent (BAW-2) or 26 percent (BWC) at the point of minimum DNBR as a basis for less than four reactor coolant pumps operating of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

A2

The DNBR as calculated by the BAW-2 or the BWC correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

The maximum thermal power, as a function of reactor coolant pump operation is limited by the power level trip produced by the flux-flow ratio (percent flow x flux-flow ratio), plus the appropriate calibration and instrumentation errors.

For each combination of operating reactor coolant pumps of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2) or 26 percent (BWC) for that particular reactor coolant pump combination. The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive because any pressure-temperature point above and to the left of this curve will be above and to the left of the other curves.

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May, 1976.

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, April, 1985.

(3) FSAR, Section 3.2.3.1.1.c.

Using a local quality limit of 22 percent (BAW-2) or 26 percent (BWC) at the point of minimum DNBR as a basis for less than four reactor coolant pumps operating of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 or the BWC correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

A2

The maximum thermal power, as a function of reactor coolant pump operation is limited by the power level trip produced by the flux-flow ratio (percent flow x flux-flow ratio), plus the appropriate calibration and instrumentation errors.

For each combination of operating reactor coolant pumps of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2) or 26 percent (BWC) for that particular reactor coolant pump combination. The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive because any pressure-temperature point above and to the left of this curve will be above and to the left of the other curves.

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May, 1976.

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, April, 1985.

(3) FSAR, Section 3.2.3.1.1.c.

2.0

(SLs)

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

A1

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

A1

Specification

2.1.2 — 2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel. MODES 1, 2, 3, 4, and 5

2.1.2 Appl

L1

<LATER> (3.4B) — 2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1968.

LATER

Bases

The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110 percent of design pressure.(2) The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110 percent of design pressure. Thus, the safety limit of 2750 psig (110 percent of the 2500 psig design pressure) has been established.(2) The settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig ±1%)(3) have been established to assure that the reactor coolant system pressure safety limit is not exceeded. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig +1, -8%. However, if found outside of a ±1% tolerance band, they shall be reset to 2500 psig ±1%. The initial hydrostatic test is conducted at 3125 psig (125 percent of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.(4)

A2

REFERENCES

(1) FSAR, Section 4

(2) FSAR, Section 4.3.11.1

(3) FSAR, Section 4.2.4

(4) FSAR, Table 4-1

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

<LATER> (2.0)

LATER

~~**Applicability**~~

~~Applies to the limit on reactor coolant system pressure.~~

~~**Objective**~~

~~To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.~~

~~**Specification**~~

~~2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.~~

3.4.10 LCO

2.2.2 ~~The setpoint of~~ the pressurizer code safety valves shall be OPERABLE ~~in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1968.~~

LA1 Bases

Bases

A2

~~The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110 percent of design pressure.(2) The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110 percent of design pressure. Thus, the safety limit of 2750 psig (110 percent of the 2500 psig design pressure) has been established. (2) The settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig ±1%)(3) have been established to assure that the reactor coolant system pressure safety limit is not exceeded. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig +1, -3%. However, if found outside of a ±1% tolerance band, they shall be reset to 2500 psig ±1%. The initial hydrostatic test is conducted at 3125 psig (125 percent of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.(4)~~

~~**REFERENCES**~~

~~(1) FSAR, Section 4~~

~~(2) FSAR, Section 4.3.11.1~~

~~(3) FSAR, Section 4.2.4~~

~~(4) FSAR, Table 4-1~~

<ADD SR 3.4.10.1 as-left lift setting criterion> A8

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

A1

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

Specification

LCO 3.3.1

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and the Protection System Maximum Allowable Setpoint for Axial Power Imbalance as given in the COLR.

LA1

Bases

Bases

A2

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in Table 2.3-1. The safety analysis has been based upon these protection system instrumentation trip setpoints plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 104.9 percent of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis.

A. Overpower Trip Based on Flow and Imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant-flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.30 (BAW-2) or 1.18 (BWC) should a low flow condition exist due to any electrical malfunction.

A2

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage. For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of the Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR are produced. The power-to-flow ratio reduces the power level trip associated reactor power-to-reactor power imbalance boundaries by the value specified in the COLR for a 1 percent flow reduction.

B. Pump Monitors

In conjunction with the power imbalance/flow trip, the pump monitors prevent the minimum core DNBR from decreasing below 1.30 (BAW-2) or 1.18 (BWC) by tripping the reactor due to the loss of reactor coolant

3.3.1

A2

pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Table 2.3-1 for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.(2)

The low pressure (1800 psig) and variable low pressure (GOLR) trip setpoint shown in Table 2.3-1 have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction.(2,3)

To account for the calibration and instrumentation errors, the accident analysis used the protective limit specified in the COLR.

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) shown in Table 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620 F.

E. Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

F. Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip set point of ≤5.0 percent of rated power is automatically imposed during reactor shutdown.
2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

A2

The purpose of the 1720 psig high-pressure trip setpoint is to prevent normal operation with part of the reactor protection system bypassed. This high-pressure trip setpoint is lower than the normal low-pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The overpower trip setpoint of ≤5.0 prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

REFERENCES

(1) FSAR, Section 14.1.2.3

(2) FSAR, Section 14.1.2.2

(3) FSAR, Section 14.1.2.7

(4) FSAR, Section 14.1.2.8

(5) FSAR, Section 14.1.2.6

A1

Table 2.3-1
Reactor Protection System Trip Setting Limits

Table 3.3.1-1 Allowable Values

<LATER> (3.4A)

LATER

Function #		Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power, 75%)	One Reactor Coolant Pump Operating in Each Loop[d] (Nominal Operating Power, 49%)	Shutdown Bypass
1.a/1.b	Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
8	Nuclear Power based on flow[b] and imbalance, % of rated, max	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Bypassed
7	Nuclear Power based on pump monitors, % of rated, max[c]	NA	NA	55	Bypassed
3/11	High RC system pressure, psig, max	2355	2355	2355	1720[a]
4	Low RC system pressure, psig. min	1800	1800	1800	Bypassed
5	Variable low RC system pressure, psig, min	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Bypassed
2	RC temp, F, max	618	618	618	618
6	High reactor building pressure, psig, max	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)

A1

A12

LA1 Bases

(a) Automatically set when other segments of the RPS (as specified) are bypassed.

(b) Reactor coolant system flow, %

(c) The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

<LATER> (3.4A)

(d) Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hrs. with the reactor critical. LATER

<Add Table 3.3.1-1 Allowable Values for Functions 9 & 10> M19

<Add Table 3.3.1-1 Note (a)> A1

3.3.1

A1

M1

LA1 Bases

3.4.4 LCO
3.4.4 RA A.1

Table 2.3-1
Reactor Protection System Trip Setting Limits

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	~~Three Reactor Coolant Pumps Operating (Nominal Operating Power, 75%)~~	~~One Reactor Coolant Pump Operating in Each Loop[d] (Nominal Operating Power, 49%)~~	Shutdown Bypass
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
Nuclear Power based on flow[b] and imbalance, % of rated, max	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Bypassed
Nuclear Power based on pump monitors, % of rated, max[c]	NA	NA	55	Bypassed
High RC system pressure, psig, max	2355	2355	2355	1720[a]
Low RC system pressure, psig. min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Bypassed
RC temp, F, max	618	618	618	618
High reactor building pressure, psig, max	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)

⟨LATER⟩ (3.3A)

–LATER

(a) Automatically set when other segments of the RPS (as specified) are bypassed.

(b) Reactor coolant system flow, %

(c) The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

3.4.4 RA A.1
RA B.1

(d) Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hrs. with the reactor critical.

M1

3.4.4

ITS

~~3. LIMITING CONDITIONS FOR OPERATION~~ (A1)

3.0 LIMITING CONDITION FOR OPERATION (LCO) ~~(GENERAL)~~ APPLICABILITY

LCO 3.0.1 3.0.1 The Limiting Conditions for Operation requirements shall be applicable during the REACTOR OPERATING CONDITIONS or other conditions ~~specif~~ied for each specification. (A6)

LCO 3.0.2 3.0.2 Adherence to the requirements of the Limiting Condition for Operation within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, no further actions need be taken. (A7)

LCO 3.0.3 3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated Action requirements, within one hour action shall be initiated to place the unit in an OPERATING CONDITION in which the Specification does not apply by placing it, as applicable, in: (M2)

1. At least HOT STANDBY within the next 6 hours,

2. At least HOT SHUTDOWN within the following 6 hours, and

3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the Action requirements, the Action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications. (LCO 3.03 is only applicable in MODES 1,2,3 and 4.) (A4)

LCO 3.0.4 3.0.4 Entry into a REACTOR OPERATING CONDITION or other specified condition shall not be made when the conditions of the Limiting Conditions for Operation are not met and the associated action requires a shutdown if they are not met within a specified time interval. Entry into a REACTOR OPERATING CONDITION or other specified condition may be made in accordance with Action requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to REACTOR OPERATING CONDITIONS as required to comply with Action requirements. Exceptions to these requirements are stated in the individual specification. (L1)

<ADD LCO 3.0.5> (A3)

<ADD LCO 3.0.6> (L2)

<ADD LCO 3.0.7> (A5)

3.0

ITS

(A1)

LIMITING CONDITION FOR OPERATION (continued)

3.0.5 When a system, subsystem, train, component or device is determined to b inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE; or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, within 2 hours action shall be initiated to place the unit in an OPERATING CONDITION in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in: — LATER

(LATER) (3.8)

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This Specification is not applicable in Cold Shutdown or Refueling Shutdown.

BASES

3.0.1 through 3.0.4 Establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shutdown the reactor or follow any remedial Action permitted by the Technical Specification until the condition can be met."

3.0.1 Establishes the Applicability statement within each individual Specification as the requirement for when (i.e., in which operational modes or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The Action requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

(A2)

There are two basic types of Action requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the Action requirements. In this case, conformance to the Action requirements provides an acceptable level of safety for unlimited continued operation as long as the Action requirements continue to be met. The second type of Action requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to

LIMITING CONDITION FOR OPERATION (continued)

3.8.1 RA A.2
3.8.1 RA B.2
3.8.1 RA C.1

L1 M16

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE; or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, ~~within 2 hours~~ action shall be initiated to place the unit in an OPERATING CONDITION in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in:

A1

1. ~~At least HOT STANDBY within the next 6 hours~~,
2. At least ~~HOT SHUTDOWN~~ MODE 3 within ~~the following 6~~ 12 hours, and (3.8.1 RA F.1)
3. At least ~~COLD SHUTDOWN~~ MODE 5 within ~~the subsequent 24~~ 36 hours. (3.8.1 RA F.2)

A3

This Specification is not applicable in ~~Cold Shutdown or Refueling Shutdown~~ MODE 5 or 6.

BASES

A2

~~3.0.1 through 3.0.4 Establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):~~

~~"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shutdown the reactor or follow any remedial Action permitted by the Technical Specification until the condition can be met."~~

~~3.0.1 Establishes the Applicability statement within each individual Specification as the requirement for when (i.e., in which operational modes or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The Action requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.~~

~~There are two basic types of Action requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the Action requirements. In this case, conformance to the Action requirements provides an acceptable level of safety for unlimited continued operation as long as the Action requirements continue to be met. The second type of Action requirement specifies a time limit in which conformance to the Action requirements of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to~~

BASES (continued)

restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these Actions are not completed withi the allowable outage time limits, a shutdown is required to place the facilit in a mode or condition in which the Specification no longer applies. It is not intended that the shutdown Action requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

The specified time limits of the Action requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the Action requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual Specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the Action requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with Action requirements, the plant may have entered a mode in which a new specification becomes applicable. In this case, the time limits of the Action requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Conditio for Operation are not met.

3.0.2 Establishes that noncompliance with a Specification exists when the requirements of the Limiting Condition for Operation are not met and the associated Action requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the Action requirements within the specified time interval constitutes compliance with a Specification and (2) completion of the remedia measures of the Action requirements is not required when compliance with a Limiting Condition for Operation is restored within the time interval specified in the associated Action requirements.

A2

3.0.3 Establishes the shutdown Action requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated Action requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown mode when plant operation cannot be maintained within the limit for safe operation defined by the Limiting Conditions for Operation and its Action requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result i redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower modes of operation permit the shutdown to proceed in a controlled and orderly

BASES (continued)

manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant systems and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the Action requirements are completed, the shutdown may be terminated. The time limits of the Action requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the Action requirements have been met or the time limits of the Action requirements have not expired, thus providing an allowance for the completion of the required Actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN condition when a shutdown is required during the POWER mode of operation. If the plant is in a lower mode of operation when a shutdown is required, the time limit for reaching the next lower mode of operation applies. However, if a lower mode of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable mode, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to reach a lower mode of operation in less than the total time allowed. A2

The same principle applies with regard to the allowable outage time limits of the Action requirements, if compliance with the Action requirements for one specification results in entry into a mode or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of Action requirements for a higher mode of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower mode of operation.

The shutdown requirements of Specification 3.0.3 do not apply in COLD SHUTDOWN and REFUELING SHUTDOWN, because the Action requirements of individual specifications define the remedial measures to be taken.

<u>3.0.4</u> Establishes limitations on mode changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher mode of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the Action requirements if a change in modes were permitted. The purpose of this specification is to ensure that facility operation is not

BASES (continued)

initiated or that higher modes of operation are not entered when corrective action is being taken to obtain compliance with a Specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with Action requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a mode change. Therefore, in this case, if the requirements for continued operation have been met in accordance with the requirements of the Specification, then entry into that mode of operation is permissible. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

A2

When a shutdown is required to comply with Action requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower mode of operation. For the purpose of compliance with this specification the term 'shutdown' is defined as a required reduction in the REACTOR OPERATING CONDITION.

3.0.5 Delineates what additional conditions must be satisfied to permit operation to continue when a normal or emergency power source is not OPERABLE It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the Limiting Condition for Operation statements associated with individual systems, subsystems, trains, components or devices to be consistent with the Limiting Condition for Operation statements of the associated electrical power source. It allows operation to be governed by the time limits of the Limiting Condition for Operation for the normal or emergency power source, not the individual Limiting Condition for Operation statements for each system, subsystem, train component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.7.2.C provides for a 7 day out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable Action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to

3.8.1

BASES (continued)

A2

initiated or that higher modes of operation are not entered when corrective action is being taken to obtain compliance with a Specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with Action requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a mode change. Therefore, in this case, if the requirements for continued operation have been met in accordance with the requirements of the specification, then entry into that mode of operation is permissible. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with Action requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower mode of operation. For the purpose of compliance with this specification the term 'shutdown' is defined as a required reduction in the REACTOR OPERATING CONDITION.

3.0.5 Delineates what additional conditions must be satisfied to permit operation to continue when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the Limiting Condition for Operation statements associated with individual systems, subsystems, trains, components or devices to be consistent with the Limiting Condition for Operation statements of the associated electrical power source. It allows operation to be governed by the time limits of the Limiting Condition for Operation for the normal or emergency power source, not the individual Limiting Condition for Operation statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.7.2.C provides for a 7 day out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable Action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to

3.8.1

A2

be consistent with the Limiting Condition for Operation statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be OPERABLE, and all redundant systems, subsystems, trains, components and devices must be OPERABLE, or otherwise satisfy Specification 3.0.5 (i.e., be capable of performing their design function and have at least one normal or one emergency power source OPERABLE). If they are not satisfied, shutdown is required in accordance with this specification.

As a further example, Specification 3.7.1.A requires in part that two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system be OPERABLE. Specification 3.7.2.B provides a 24 hour out-of-service time when both required offsite circuits are not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable normal power sources, both of the offsite circuits would also be inoperable. This would dictate invoking the applicable Limiting Condition for Operation statements for each of the applicable LCOs. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the Limiting Condition for Operation statement for the inoperable normal power sources instead, provided the other specified conditions are satisfied. In this case, this would mean that for one division the emergency power source must be OPERABLE (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other division must be OPERABLE, or likewise satisfy Specification 3.0.5 (i.e., be capable of performing their design functions and have an emergency power source OPERABLE). In other words, both emergency power sources must be OPERABLE and all redundant systems, subsystems, trains, components and devices in both divisions must also be OPERABLE. If these conditions are not satisfied, shutdown is required in accordance with this specification.

During Cold Shutdown and Refueling Shutdown, Specification 3.0.5 is not applicable and thus the individual Action statements for each applicable Limiting Condition for Operation in these MODES must be adhered to.

3.4.4 3.4.7
3.4.5 3.4.8
3.4.6

<Add SR 3.4.4.1> M3

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.

3.1.1 Operational Components

Specification

A1

A17

3.1.1.1 Reactor Coolant Pumps <Add 3.4.4 RA B.1.>

3.4.4 LCO & Appl.
3.4.4 RA A.1

A. Pump combinations permissible for given power levels shall be as shown in Table 2.3-1. Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hours with the reactor critical. MODES 1 & 2 M1

3.4.5 LCO Note a
3.4.5 RA C.1
3.4.6 RA B.1
3.4.6/7/8 LCO Note 1 a
3.4.7 RA B.1
3.4.8 RA B.1
& <LATER> (3.9)

B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant. With no reactor coolant pumps or decay heat removal pumps running, ~~immediately suspend all operations involving a~~ reduction of boron concentration in the reactor coolant system. A3

LA1 Bases

3.1.1.2 Steam Generator

3.4.4 LCO & Appl.
3.4.5 LCO & Appl.

A. Two steam generators shall be operable whenever the reactor coolant average temperature is above 280°F. MODES 1, 2 & 3 A1

3.1.1.3 Pressurizer Safety Valves

<LATER> (3.4B)

LATER

A. Both pressurizer code safety valves shall be operable when the reactor is critical. With one pressurizer code safety valve inoperable, either restore the valve to operable status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

B. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. The provisions of Specification 3.0.3 are not applicable.

3.1.1.4 Reactor Internals Vent Valves

The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1. The provisions of Specification 3.0.3 are not applicable. LA3 TRM

3.1.1.5 Reactor Coolant Loops MODE 1, 2, 3 A4

3.4.4 LCO & Appl.
3.4.5 LCO & Appl.

A. With the reactor coolant average temperature above 280°F, the reactor coolant loops listed below shall be operable:

3.9.4
3.9.5

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.

<LATER> (3.4A)

3.1.1 Operational Components

Specification

3.1.1.1 Reactor Coolant Pumps

A. Pump combinations permissible for given power levels shall be as shown in Table 2.3-1. Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hours with the reactor critical.

LATER

B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant. With no ~~reactor coolant pumps or~~ decay heat removal pumps running, immediately suspend all operations involving a reduction of boron concentration in the reactor coolant system.

LA1 Bases

LATER

3.9.4 RA A.1

& <LATER> (3.4A)

3.1.1.2 Steam Generator

A. Two steam generators shall be operable whenever the reactor coolant average temperature is above 280°F.

<LATER> (3.4A) — LATER

3.1.1.3 Pressurizer Safety Valves

A. Both pressurizer code safety valves shall be operable when the reactor is critical. With one pressurizer code safety valve inoperable, either restore the valve to operable status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

B. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. The provisions of Specification 3.0.3 are not applicable.

<LATER> (3.4B) — LATER

3.1.1.4 Reactor Internals Vent Valves

The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1. The provisions of Specification 3.0.3 are not applicable.

<LATER> (3.4A) — LATER

3.1.1.5 Reactor Coolant Loops

A. With the reactor coolant average temperature above 280°F, the reactor coolant loops listed below shall be operable:

<LATER> (3.4A) — LATER

<Add 3.4.10 LCO Note 2 & SR 3.4.10.1 Note> L17

A1

3.1 ~~REACTOR COOLANT SYSTEM~~

~~Applicability~~

~~Applies to the operating status of the reactor coolant system.~~

~~Objective~~

~~To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.~~

3.1.1 ~~Operational Components~~

~~Specification~~

<LATER> (3.4A) 3.1.1.1 ~~Reactor Coolant Pumps~~ LATER

A. ~~Pump combinations permissible for given power levels shall be as shown in Table 2.3-1. Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hours with the reactor critical.~~

<LATER> (3.4A & 3.9) B. ~~The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant. With no reactor coolant pumps or decay heat removal pumps running, immediately suspend all operations involving a reduction of boron concentration in the reactor coolant system.~~ LATER

<LATER> (3.4 A) 3.1.1.2 ~~Steam Generator~~ LATER

A. ~~Two steam generators shall be operable whenever the reactor coolant average temperature is above 280°F.~~

3.1.1.3 Pressurizer Safety Valves

3.4.10 LCO & APPL
3.4.10 RA A.1
3.4.10 RA B.1

A. Both pressurizer code safety valves in MODES 1 & 2 shall be operable ~~when the reactor is critical.~~ With one pressurizer code safety valve inoperable, either restore the valve to operable status within 15 minutes or be in ~~HOT~~ MODE 3 ~~SHUTDOWN~~ within ~~12~~ 6 hours. M2 A1

3.4.10 LCO Note 1

B. When the reactor is ~~subcritical~~ in MODE 3 or MODE 4 @ > LTOP enable temp at least one pressurizer code safety valve shall be operable ~~if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. The provisions of Specification 3.0.3 are not applicable.~~ L15

<LATER> (3.4A) 3.1.1.4 ~~Reactor Internals Vent Valves~~ LATER

~~The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1. The provisions of Specification 3.0.3 are not applicable.~~

<LATER> (3.4A) 3.1.1.5 ~~Reactor Coolant Loops~~ LATER

A. ~~With the reactor coolant average temperature above 280°F, the reactor coolant loops listed below shall be operable:~~

<Add 3.4.10 RA C.1 & Cond.B - second entry condition> M2

1.1

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

(LATER) (3.4A)

Otherwise, restore the required loops to operable status within 72 hours or reduce the reactor coolant average temperature to less than or equal to 280°F within the next 12 hours.

B. With the reactor coolant average temperature above 280°F, at least one of the reactor coolant loops listed above shall be in operation.

Otherwise, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

3.1.1.6 Decay Heat Removal

With the reactor coolant average temperature at or below 280°F, but the reactor above the refueling shutdown condition, at least two of the coolant loops listed below shall be operable, and at least one loop shall be in operation:*

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump.

3. Decay Heat Removal Loop (A)**

4. Decay Heat Removal Loop (B)**

A8

A. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

B. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*All reactor coolant pumps and decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**The normal or emergency power source may be inoperable when the reactor is in a cold shutdown condition.

A8

3.4.5
3.4.6

L7

<Add 3.4.5 LCO Note b> M15

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump. LA1 Bases

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

3.4.5 RA A.1 / RA B.1 — Otherwise, restore the required loops to operable status within 72 hours or reduce the reactor coolant average temperature to less than or equal to 280°F within the next 12 hours. A3

MODE 4 — In MODE 3 — A1

3.4.5 LCO & Note a — B. With the reactor coolant average temperature above 280°F, at least one of the reactor coolant loops listed above shall be in operation.

3.4.5 RA C.1 / RA C.2 — Otherwise, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation. L7

3.1.1.6 Decay Heat Removal

In MODES 4 and 5 — A12

3.4.6 LCO & Appl. [3.4.7 LCO] [See page 16a-2] [3.4.8 LCO] [See page 16a-3] — With the reactor coolant average temperature at or below 280°F, but the reactor above the refueling shutdown condition, at least two of the coolant loops listed below shall be operable, and at least one loop shall be in operation:*

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump. LA1 Bases

2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump.

3. Decay Heat Removal Loop (A)**

4. Decay Heat Removal Loop (B)**

3.4.6 RA A.1 / RA B.2 / RA A.2 — A. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours. L1

3.4.6 RA B.1 / RA B.2 — B. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

removed from operation — A14

3.4.6 LCO Note — *All reactor coolant pumps and decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

<LATER> (1.0) — **The normal or emergency power source may be inoperable when the reactor is in a cold shutdown condition. — LATER

<Add 3.4.7 LCO Notes 2 & 3> L3

[3.4.5]
[See page 16a-1]

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

Otherwise, restore the required loops to operable status within 72 hours or reduce the reactor coolant average temperature to less than or equal to 280°F within the next 12 hours.

B. With the reactor coolant average temperature above 280°F, at least one of the reactor coolant loops listed above shall be in operation.

Otherwise, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

3.4.7 LCO & App.
[3.4.6 LCO]
[See page 16a-1]
[3.4.8 LCO]
[See page 16a-3]

3.1.1.6 Decay Heat Removal

In MODES 4 & 5

~~With the reactor coolant average temperature at or below 280°F, but the reactor above the refueling shutdown condition,~~ at least two of the coolant loops listed below shall be operable, and at least one loop shall be in operation:* A12 M4

1. ~~Reactor Coolant Loop (A) and its associated~~ steam generator and at least one associated reactor coolant pump. M4 L2

2. ~~Reactor Coolant Loop (B) and its associated~~ steam generator and at least one associated reactor coolant pump. LA1 Bases

3. Decay Heat Removal Loop (A)**

4. Decay Heat Removal Loop (B)**

3.4.7 RA A.1/A.2 A. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours. A5
RA B.2

B. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.
3.4.7 RA B.1
B.2

removed from operation A14

3.4.7 LCO Note 1 *All reactor coolant pumps and decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

<LATER> (1.0) ~~**The normal or emergency power source may be inoperable when the reactor is in a cold shutdown condition.~~ LATER

<Add 3.4.8, RA B.2> M16 3.4.8

<Add 3.4.8 LCO Note 1.b> M12

<Add 3.4.8 LCO Note 2> L3

[3.4.5] [See page 16a-1]

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

Otherwise, restore the required loops to operable status within 72 hours or reduce the reactor coolant average temperature to less than or equal to 280°F within the next 12 hours.

B. With the reactor coolant average temperature above 280°F, at least one of the reactor coolant loops listed above shall be in operation.

Otherwise, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

3.1.1.6 Decay Heat Removal

3.4.8 LCO & Appl. / 3.4.6 LCO [See page 16a-1] / 3.4.7 LCO [See page 16a-2]

In MODES 4 & 5 ~~With the reactor coolant average temperature at or below 280°F, but the reactor above the refueling shutdown condition,~~ at least two of the coolant loops listed below shall be operable, and at least one loop shall be in operation:* — A12

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump. — L2

3. Decay Heat Removal Loop (A)**

4. Decay Heat Removal Loop (B)**

3.4.8 RA A.1 RA B.3

A. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible; ~~be in COLD SHUTDOWN within 20 hours.~~ — A5

3.4.8 RA B.1/B.3

B. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

3.4.8 LCO Note 1.a

*All reactor coolant pumps and decay heat removal pumps may be ~~de-energized~~ removed from operation for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, ~~and (2) core outlet temperature is maintained at least 10°F below saturation temperature.~~ — A14 — L4

<LATER> (1.0) ~~**The normal or emergency power sources may be inoperable when the reactor is in a cold shutdown condition.~~ — LATER

3.4A

3.1.1.7 Reactor Coolant System Vents

At least one reactor coolant system vent path consisting of at least two valves in series shall be operable at each of the following locations whenever the Reactor Coolant average temperature is above 280F.

1. Reactor vessel head

2. Pressurizer steam space

3. Reactor coolant system Hot Leg high point (2 locations)

LA2

TRM

A. With one of the above vent paths inoperable, startup and/or power operation may continue provided the inoperable vent path is maintained closed; restore the inoperable vent path to operable status within 30 days, or be in hot standby within 6 hours and in cold shutdown within the following 30 hours.

L9

B. With two or more of the above vent paths inoperable, maintain the inoperable vent paths closed and restore at least two vent paths to operable status within 72 hours or be in hot standby within 6 hours and in cold shutdown within the following 30 hours.

3.4.4
3.4.5
3.4.6
3.4.7
3.4.8

BASES:

The plant is designed to operate with both reactor coolant loops and at least one reactor coolant pump per loop in operation, and maintain DNBR above 1.30 (for the BAW-2 correlation) and 1.18 (for the BWC correlation) during all normal operations and anticipated transients.(1)

(A2)

Whenever the reactor coolant average temperature is above 280°F, single failure considerations require that two loops be operable.

The decay heat removal system suction piping is designed for 300°F thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident. (5) The pressurizer code safety valve lift setpoint shall be 2,500 psig ±1 percent allowance for error and each valve shall be capable of relieving 300,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig +1, -3 percent. However, if found outside the ±1 percent tolerance band, they shall be reset to 2500 psig ±1 percent.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internal vent valves (1) ensure operability, (2) ensure that the valves are not open during normal operation, and (3) demonstrate that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

The reactor coolant 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 coolant system vent path from the reactor vessel head, the reactor coolant system highpoints, and the pressurizer steam space ensures the capability exists to perform this function. The valve redundancy of the vent paths serves to minimize the probability of inadvertent actuation and breach of reactor coolant pressure boundary while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. Testing requirements are covered in Section 4.0 for the class 2 valves and Table 4.1-2 for the vent paths. These are consistent with ASME Section XI and Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," 11/80.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7
(2) FSAR, Section 4.2.5.1 and 9.5.2.3
(3) FSAR, Section 4.2.5.4
(4) FSAR, Section 4.3.10.4 and 4.2.4
(5) FSAR, Section 4.3.7

3.4.10

BASES:

The plant is designed to operate with both reactor coolant loops and at least one reactor coolant pump per loop in operation, and maintain DNBR above 1.30 (for the BAW-2 correlation) and 1.18 (for the BWC correlation) during all normal operations and anticipated transients. (1)

Whenever the reactor coolant average temperature is above 280°F, single failure considerations require that two loops be operable.

A2

The decay heat removal system suction piping is designed for 300°F thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident. (5) The pressurizer code safety valve lift setpoint shall be 2,500 psig ±1 percent allowance for error and each valve shall be capable of relieving 300,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig +1, -3 percent. However, if found outside the ±1 percent tolerance band, they shall be reset to 2500 psig ±1 percent.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internal vent valves (1) ensure operability, (2) ensure that the valves are not open during normal operation, and (3) demonstrate that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

The reactor coolant 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 coolant system vent path from the reactor vessel head, the reactor coolant system highpoints, and the pressurizer steam space ensures the capability exists to perform this function. The valve redundancy of the vent paths serves to minimize the probability of inadvertent actuation and breach of reactor coolant pressure boundary while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. Testing requirements are covered in Section 4.0 for the class 2 valves and Table 4.1-2 for the vent paths. These are consistent with ASME Section XI and Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," 11/80.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7
(2) FSAR, Section 4.2.5.1 and 9.5.2.3
(3) FSAR, Section 4.2.5.4
(4) FSAR, Section 4.3.10.4 and 4.2.4
(5) FSAR, Section 4.3.7

3.4.3

< Add 3.4.3 Appl. > A15

3.1.2 Pressurization, Heatup, and Cooldown Limitations

Specification

3.4.3 LCO 3.1.2.1 Hydro Tests

SR 3.4.3.1
SR 3.4.3.2
SR 3.4.3.3 & Note

For thermal steady state system hydro tests, the system may be pressurized to the limits set forth in Specification 2.2 when there are fuel assemblies in the core, under the provisions of 3.1.2.3, and to ASME Code limits when no fuel assemblies are present provide the reactor coolant system limits are to the right of and below the limit line in Figure 3.1.2-1. ~~The provisions of Specifications 3.0.3 are not applicable.~~ M7 A6

~~3.1.2.2 Leak Tests~~

~~Leak tests required by Specification 4.3 shall be conducted under the provision of 3.1.2.3. The provisions of Specification 3.0.3 ar not applicable.~~ A11 M2

3.4.3 LCO & Note 3.1.2.3 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-2 and Figure 3.1.2-3, and are as follows:

Heatup:

SR 3.4.3.1 & Note
SR 3.4.3.4 & Note

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1.2-2. The heatup rates shall not exceed those shown in Figure 3.1.2-2. A16

SR 3.4.3.2 & Note

Cooldown:

Allowable combinations of pressure and temperature for a specific cooldown shall be to the right of and below the limit line in Figure 3.1.2-3. Cooldown rates shall not exceed those shown in Figure 3.1.2-3. M7

~~3.1.2.4 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100F.~~ R TRM

~~3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430F.~~ R TRM

3.1.2.6 With the limits of Specifications 3.1.2.3 ~~or 3.1.2.4 or 3.1.2.5~~ exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least ~~HOT STANDBY~~ MODE 3 within the next 6 hours and ~~reduce the RCS Tavg to less than 200F,~~ Be in MODE 5 while maintaining RCS temperature and pressure below the curve, within the following 30 hours. R TRM LA1 Bases M5 M6 A1 A7

3.4.3 RA A.1
3.4.3 RA A.2
3.4.3 RA B.1
3.4.3 RA B.2

Items on this page also addressed in the following packages: 3,4 A

SR

3.1.2 Pressurization, Heatup, and Cooldown Limitations

Specification

3.1.2.1 Hydro Tests

For thermal steady state system hydro tests, the system may be pressurized to the limits set forth in Specification 2.2 when there are fuel assemblies in the core, under the provisions of 3.1.2.3, and to ASME Code limits when no fuel assemblies are present provided the reactor coolant system limits are to the right of and below the limit line in Figure 3.1.2-1. The provisions of Specifications 3.0.3 are not applicable.

3.1.2.2 Leak Tests

Leak tests required by Specification 4.3 shall be conducted under the provision of 3.1.2.3. The provisions of Specification 3.0.3 are not applicable.

3.1.2.3 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-2 and Figure 3.1.2-3, and are as follows:

Heatup:

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1.2-2. The heatup rates shall not exceed those shown in Figure 3.1.2-2.

Cooldown:

Allowable combinations of pressure and temperature for a specific cooldown shall be to the right of and below the limit line in Figure 3.1.2-3. Cooldown rates shall not exceed those shown in Figure 3.1.2-3.

~~3.1.2.4 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100F.~~ R TRM

~~3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430F.~~ R TRM

~~3.1.2.6 With the limits of Specifications 3.1.2.3 or 3.1.2.4 or 3.1.2.5 exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tavg to less than 200F, while maintaining RCS temperature and pressure below the curve, within the following 30 hours.~~ R TRM

3.1.2.7 Prior to reaching thirty one effective full power years of operation, Figures 3.1.2-1, 3.1.2-2 and 3.1.2-3 shall be updated for the next service period in accordance with 10CFR50, Appendix G. The service period shall be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with the latest revision of Topical Report BAW-1543(5). The highest predicted adjusted reference temperature of all the beltline region materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.8. The provisions of Specification 3.0.3 are not applicable.

LA1

Bases

3.1.2.8 The updated proposed technical specifications referred to in 3.1.2.7 shall be submitted for NRC review at least 90 days prior to the end of the service period.

LA1

Bases

<LATER>

(3.4B)

3.1.2.9 With the exception of ASME Section XI testing and when the core flood tank is depressurized, during a plant cooldown the core flood tank discharge valves shall be closed and the circuit breakers for the motor operators opened before depressurizing the reactor coolant system below 600 psig.

LATER

3.1.2.10 With the exception of ASME Section XI testing, fill and vent of the reactor coolant system, emergency RCS makeup and to allow maintenance of the valves, when the reactor coolant temperature is less than 262°F, the High Pressure Injection motor operated valves shall be closed with their opening control circuits for the motor operators disabled.

3.1.2.11 The plant shall not be operated in a water solid condition when the RCS pressure boundary is intact except as allowed by Emergency Operating Procedures and during System Hydrotest.

<Add 3.4.3 Condition A Note> M5

<Add 3.4.3 Condition C with Note> M5

⟨LATER⟩ (3.4A)

3.1.2.7 Prior to reaching thirty one effective full power years of operation, Figures 3.1.2-1, 3.1.2-2 and 3.1.2-3 shall be updated for the next service period in accordance with 10CFR50, Appendix G. The service period shall be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with the latest revision of Topical Report BAW-1543(5). The highest predicted adjusted reference temperature of all the beltline region materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.8. The provisions of Specification 3.0.3 are not applicable.

3.1.2.8 The updated proposed technical specifications referred to in 3.1.2.7 shall be submitted for NRC review at least 90 days prior to the end of the service period.

LATER

M7

3.4.11 LCO c w/ Note

3.1.2.9 With the exception of ASME Section XI testing and when the core flood tank is depressurized, ~~during a plant cooldown~~ the core flood tank discharge valves shall be closed ~~and the circuit breakers for the motor operators opened before depressurizing the reactor coolant system below 600 psig.~~ isolated.

LA1 Bases

L2

3.4.11 LCO b w/ Notes 1-4
3.4.11 APPL

3.1.2.10 With the exception of ASME Section XI testing, fill and vent of the reactor coolant system, emergency RCS makeup and to allow maintenance of the valves, when the reactor coolant temperature is ≤ ~~less than~~ 262°F, the High Pressure Injection ~~motor operated valves~~ shall be closed ~~with their opening control circuits for the motor operators disabled.~~ deactivated.

A12

LA1 Bases

3.4.11 LCO a w/Notes 1-3

3.1.2.11 The plant shall not be operated in a water solid condition when the RCS pressure boundary is intact except as allowed by Emergency Operating Procedures and during System Hydrotest.

A12

⟨Add 3.4.11 LCO d⟩ A12

⟨Add 3.4.11 Appl⟩ A12

⟨Add 3.4.11 ACTIONS⟩ M3

Add SR 3.4.11.1 with Notes 1-3
Add SR 3.4.11.2 with Notes 1-4
Add SR 3.4.11.3 with Note
Add SR 3.4.11.4 with Note
Add SR 3.4.11.6 with Note

M3

BASES

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.[1] These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation.[2] The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the DTT.[3]

(R) TRM

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in FTI Document 77-1258569-01 [4]. The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543. [5] The chemical composition of the limiting weld material is reported in the B&W Report, BAW-2121P [6]. The effect of neutron irradiation on the RT_{NDT} of the limiting weld material is reported in FTI Calculations 32-1245917-00 and 32-1257716-00 [7].

(A2)

Figures 3.1.2-1, 3.1.2-2, and 3.1.2-3 present the pressure-temperature limit curves for hydrostatic test, normal heatup, and normal cooldown respectively. The limit curves are applicable through the thirty first effective full power year of operation. The service period was reduced by one effective full power year from that assumed in FTI Document 77-1258569-01 to be conservative with respect to independent calculations performed by the NRC staff. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all allowed operating reactor coolant pump combinations.

The pressure-temperature limit lines shown on Figure 3.1.2-2 for reactor criticality and on Figure 3.1.2-1 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10CFR50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region.

The spray temperature difference restriction based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

(R) TRM

BASES

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.[1] These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation.[2] The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the DTT.[3]

R TRM

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in FTI Document 77-1258569-01 [4]. The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543. [5] The chemical composition of the limiting weld material is reported in the B&W Report, BAW-2121P [6]. The effect of neutron irradiation on the RT_{NDT} of the limiting weld material is reported in FTI Calculations 32-1245917-00 and 32-1257716-00 [7].

A2

Figures 3.1.2-1, 3.1.2-2, and 3.1.2-3 present the pressure-temperature limit curves for hydrostatic test, normal heatup, and normal cooldown respectively. The limit curves are applicable through the thirty first effective full power year of operation. The service period was reduced by one effective full power year from that assumed in FTI Document 77-1258569-01 to be conservative with respect to independent calculations performed by the NRC staff. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all allowed operating reactor coolant pump combinations.

The pressure-temperature limit lines shown on Figure 3.1.2-2 for reactor criticality and on Figure 3.1.2-1 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10CFR50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region.

The spray temperature difference restriction based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

R TRM

Items on this page also addressed in the following packages: 3.4A, 3.4B

SR

BASES

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.[1] These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation.[2] The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the DTT.[3]

R TRM

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in FTI Document 77-1258569-01 [4]. The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543. [5] The chemical composition of the limiting weld material is reported in the B&W Report, BAW-2121P [6]. The effect of neutron irradiation on the RT_{NDT} of the limiting weld material is reported in FTI Calculations 32-1245917-00 and 32-1257716-00 [7].

Figures 3.1.2-1, 3.1.2-2, and 3.1.2-3 present the pressure-temperature limit curves for hydrostatic test, normal heatup, and normal cooldown respectively. The limit curves are applicable through the thirty first effective full power year of operation. The service period was reduced by one effective full power year from that assumed in FTI Document 77-1258569-01 to be conservative with respect to independent calculations performed by the NRC staff. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all allowed operating reactor coolant pump combinations.

The pressure-temperature limit lines shown on Figure 3.1.2-2 for reactor criticality and on Figure 3.1.2-1 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10CFR50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region.

The spray temperature difference restriction based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

R TRM

3.4.3

The heatup and cooldown rates stated in this specification are intended as the maximum changes in temperature in one direction in a one hour period. The actual temperature linear ramp rate may exceed the stated limits for a time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the one hour period.

Specification 3.1.2.9 is to ensure that the core flood tanks are not the source for pressurizing the reactor coolant system when in cold shutdown.

Specification 3.1.2.10 is to ensure that high pressure injection is not the source of pressurizing the reactor coolant system when in cold shutdown. The LTOP enable temperature has been calculated in accordance with Code Case N-514. Instrument error is not included in the reactor coolant temperature of 262°F.

Specification 3.1.2.11 is to ensure that the reactor coolant system is not operated in a manner which would allow overpressurization due to a temperature transient.

A2

REFERENCES

(1) FSAR, Section 4.1.2.4

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) FSAR, Section 4.3.11.5

R TRM

(4) FTI Document Number 77-1258569-01

(5) BAW-1543, latest revision

(6) BAW-2121P

(7) FTI Calculation Numbers 32-1245917-00 and 32-1257716-00

A2

3.4.11

The heatup and cooldown rates stated in this specification are intended as the maximum changes in temperature in one direction in a one hour period. The actual temperature linear ramp rate may exceed the stated limits for a time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the one hour period.

Specification 3.1.2.9 is to ensure that the core flood tanks are not the source for pressurizing the reactor coolant system when in cold shutdown.

Specification 3.1.2.10 is to ensure that high pressure injection is not the source of pressurizing the reactor coolant system when in cold shutdown. The LTOP enable temperature has been calculated in accordance with Code Case N-514. Instrument error is not included in the reactor coolant temperature of 262°F.

Specification 3.1.2.11 is to ensure that the reactor coolant system is not operated in a manner which would allow overpressurization due to a temperature transient.

A2

REFERENCES

(1) FSAR, Section 4.1.2.4

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) FSAR, Section 4.3.11.5

R TRM

(4) FTI Document Number 77-1258569-01

(5) BAW-1543, latest revision

(6) BAW-2121P

(7) FTI Calculation Numbers 32-1245917-00 and 32-1257716-00

A2

Items on this page also addressed in the following packages: 3.4A , 3.4B

SR

The heatup and cooldown rates stated in this specification are intended as the maximum changes in temperature in one direction in a one hour period. The actual temperature linear ramp rate may exceed the stated limits for a time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the one hour period.

Specification 3.1.2.9 is to ensure that the core flood tanks are not the source for pressurizing the reactor coolant system when in cold shutdown.

Specification 3.1.2.10 is to ensure that high pressure injection is not the source of pressurizing the reactor coolant system when in cold shutdown. The LTOP enable temperature has been calculated in accordance with Code Case N-514. Instrument error is not included in the reactor coolant temperature of 262°F.

Specification 3.1.2.11 is to ensure that the reactor coolant system is not operated in a manner which would allow overpressurization due to a temperature transient.

REFERENCES

(1) ~~FSAR, Section 4.1.2.4~~

(2) ~~ASME Boiler and Pressure Code, Section III, N-415~~ R TRM

(3) ~~FSAR, Section 4.3.11.5~~

(4) FTI Document Number 77-1258569-01

(5) BAW-1543, latest revision

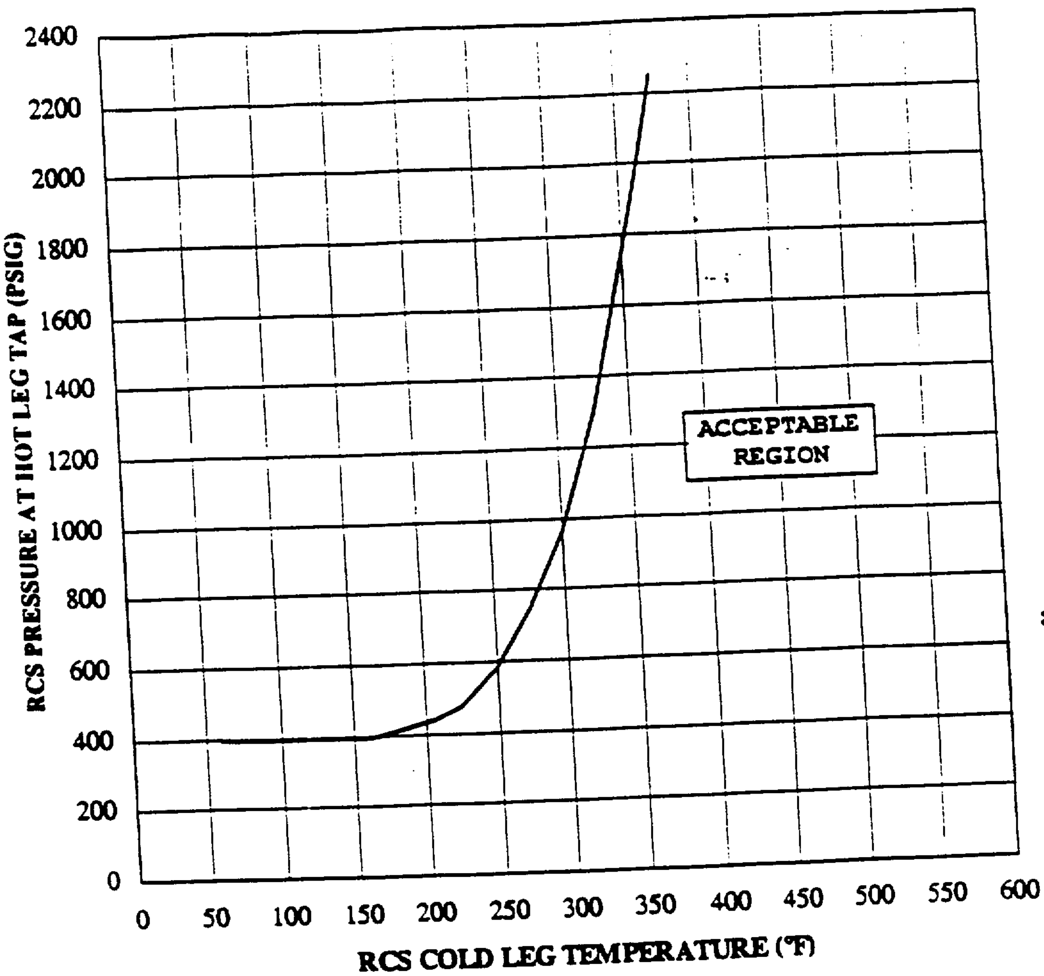
(6) BAW-2121P

(7) FTI Calculation Numbers 32-1245917-00 and 32-1257716-00

Fig 3.4.3 -3

FIGURE 3.1.2-1

RCS INSERVICE HYDROSTATIC TEST H/U & C/D LIMITS TO 31 EFPY

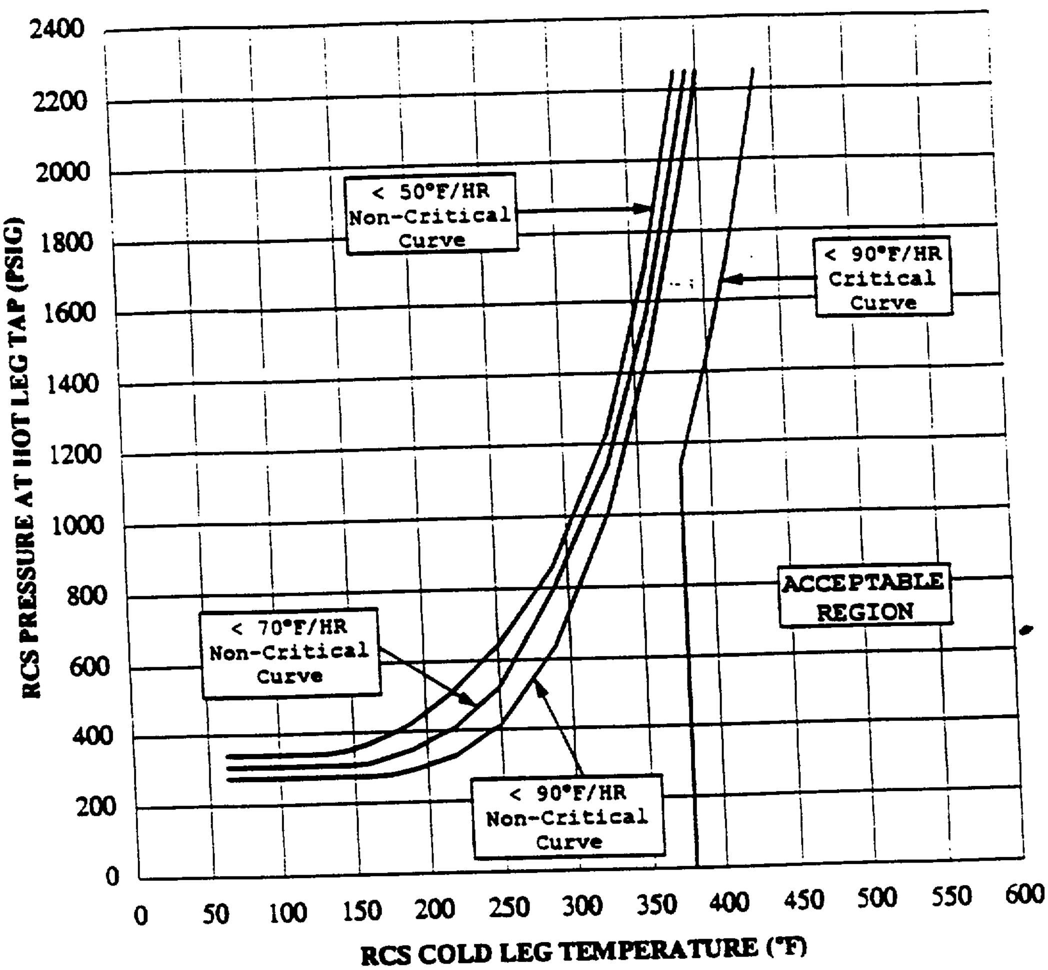


Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.
2. All Notes on Figure 3.1.2-2 are applicable for heatups. This curve is based on a heatup rate of < 90°F/HR.
3. All Notes on Figure 3.1.2-3 are applicable for cooldowns.

Fig. 3.4.3-1

FIGURE 3.1.2-2
RCS HEATUP LIMITATIONS TO 31 EFPY



Notes:

1. These curves are not adjusted for instrument error and shall not be used for operation.
2. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

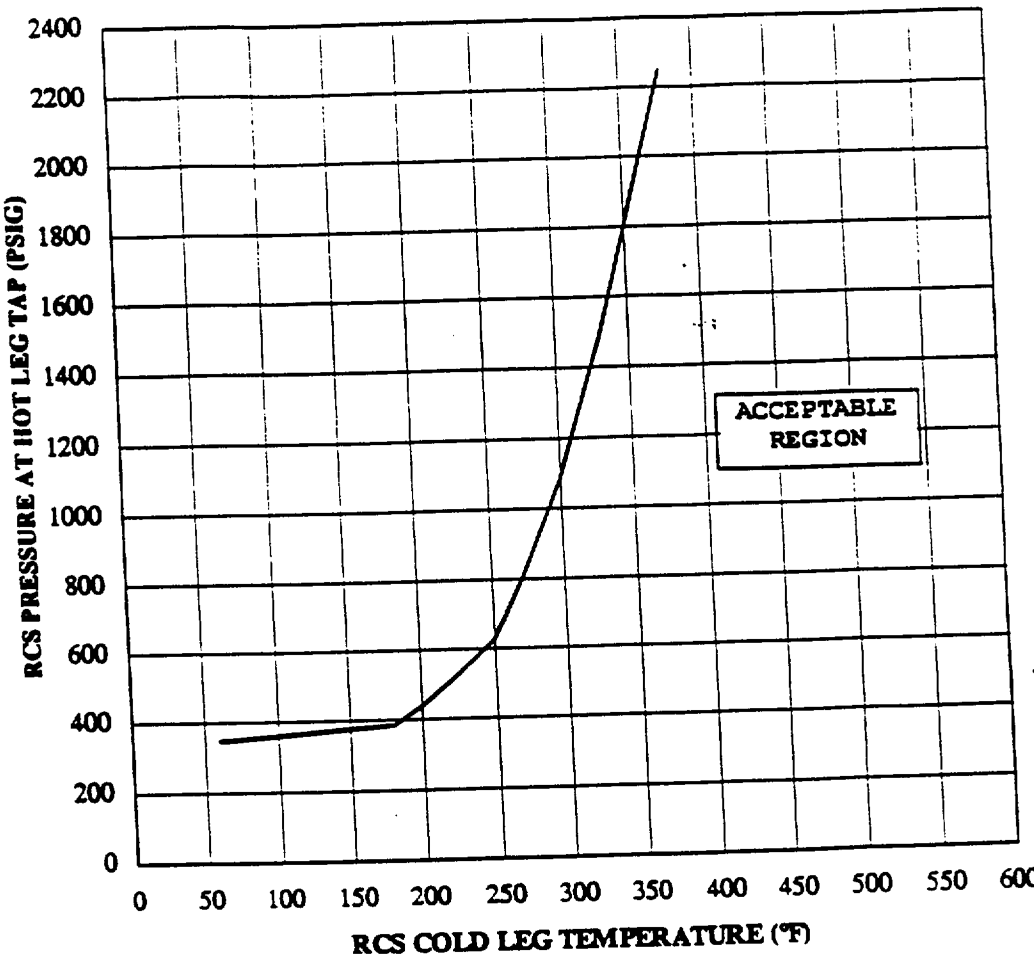
RCS TEMP	RCP RESTRICTIONS
T > 300°F	None
300°F ≥ T ≥ 225°F	≤ 3
225°F > T ≥ 84°F	≤ 2
T < 84°F	No RCPs operating

4. Allowable Heatup Rates:

RCS TEMP	H/U RATE
60°F < T ≤ 84°F	≤ 15°F/HR
T > 84°F	As allowed by applicable curve

Fig. 3.4.3-2

FIGURE 3.1.2-3
RCS COOLDOWN LIMITS TO 31 EFPY



Notes:
1. This curve is not adjusted for instrument error and shall not be used for operation.
2. A maximum step temperature change of 25°F is allowable when securing all RCPs with the DHR system in operation. This change is defined as the RCS temperature prior to securing all the RCPs minus the DHR return temperature after the RCPs are secured. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 255°F	None
150°F ≤ T ≤ 255°F	≤ 2 (See Note 5)
T < 150°F	No RCPs operating

4. Allowable Cooldown Rates:

RCS TEMP	C/D RATE	STEP CHANGE
T ≥ 280°F	100°F/HR	≤ 50°F in any 1/2 HR
280°F > T ≥ 150°F	50°F/HR (See Note 5)	≤ 25°F in any 1/2 HR
T < 150°F	25°F/HR	≤ 25°F in any 1 HR

5. If RCPs are operated < 200°F, then the RCS cooldown rate from 150°F ≤ T ≤ 180°F is reduced to 30°F in 15 hours.

3.1.5
3.1.8
3.1.9

<LATER> (3.4A) **3.1.3 Minimum Conditions for Criticality** LATER

Specification

<LATER> (3.4A 3.1.3.1 The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply. LATER A7

3.1.9 LCO

<LATER> (3.4A) 3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure 3.1.2-2. LATER

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization. M6

<LATER> (3.4B)

<LATER> (3.2) 3.1.3.4 The reactor shall be maintained subcritical by at least 1 percent Δk/k until a steam bubble is formed and an indicated water level between 45 and 305 inches is established in the pressurizer. LATER

3.1.8, 3.1.9 LCO — LATER

3.1.5 LCO, <LATER> (3.2), 3.1.5 Appl

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality MODES 1 and 2 A10 LATER A10

<LATER> (3.4B) 3.1.3.6 The reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown within the following 12 hours. LATER

3.1.5 RA A.2

<LATER> (3.4A&B) 3.1.3.7 With any ~~of the above limits violated~~ Safety rod not fully withdrawn, restore the ~~reactor~~ Safety rod to within the limit in ~~15 minutes~~ 1 hour or declare the safety rod inoperable ~~or be in at least Hot Shutdown within the next 15 minutes.~~ L6 + LATER

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests. A2

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated Δk/k would limit the magnitude of power excursion resulting from a reduction of moderator density.

<LATER> (3.4A) 3.1.3 Minimum Conditions for Criticality — LATER

Specification

<LATER> (3.4A) 3.1.3.1 The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply. — LATER
<LATER> (3.1) — LATER

<LATER> (3.4A) 3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure 3.1.2-2. — LATE

<LATER> (3.1) 3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization. — LATE

<LATER> (3.4B) 3.1.3.4 The reactor shall be maintained subcritical by at least 1 percent Δk/k until a steam bubble is formed and an indicated water level between 45 and 305 inches is established in the pressurizer. — LATER
3.2.1 App D.1

<LATER> (3.1) 3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. MODES 1+2 — LATER (A8) + LATER
3.2.1 App LATER (3.1)

<LATER> (3.4B) 3.1.3.6 The reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown within the following 12 hours. — LATER

<LATER> (3.1.3.4A-4B) 3.1.3.7 With any of the above limits violated, restore the reactor to within the limit in 15 minutes or be in at least Hot Shutdown within the next 15 minutes. — LATER

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests. (A2)

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated Δk/k would limit the magnitude of power excursion resulting from a reduction of moderator density.

<ADD SR 3.4.2.1> M14

3.4.2 APPL <LATER> (3.1) 3.1.3 Minimum Conditions for Criticality MODES 1+2 M11

Specification

≥ A8

3.4.2 LCO <LATER> (3.1) 3.1.3.1 The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply. LATER

[See Page 21-2] 3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure 3.1.2-2.

<LATER> (3.1) 3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization. LATER

<LATER> (3.4B) 3.1.3.4 The reactor shall be maintained subcritical by at least 1 percent Δk/k until a steam bubble is formed and an indicated water level between 45 and 305 inches is established in the pressurizer. LATER

<LATER> (3.1, 3.2) 3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. LATER

<LATER> (3.4B) 3.1.3.6 The reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown within the following 12 hours. LATER

3.4.2 RA A.1 + <LATER> (3.1, 3.4B) 3.1.3.7 With any of the above limits violated, restore the reactor to within the limit in 15 minutes or be in at least Hot Shutdown within the next 15 minutes. A9 + LATER

MODE 3 A1

[Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated Δk/k would limit the magnitude of power excursion resulting from a reduction of moderator density.

A2

[See Page 21-1]

3.1.3 Minimum Conditions for Criticality

Specification

SR 3.4.3.4 NOTE

3.1.3.1 The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply. A16

3.4.3 LCO
SR 3.4.3.4

3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure ~~3.1.2-2~~ 3.4.3-1

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

[See Page 21-1]

3.1.3.4 The reactor shall be maintained subcritical by at least 1 percent Δk/k until a steam bubble is formed and an indicated water level between 45 and 305 inches is established in the pressurizer.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality.

3.1.3.6 The reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown within the following 12 hours.

A1

3.4.3
RA A.1/B.1

3.1.3.7 With any of the above limits violated, restore the reactor to within the limit in ~~15~~ 30 minutes or be in ~~at least Hot Shutdown~~ MODE 3 within the next ~~15 minutes~~ 6 hours L5

Bases

<ADD 3.4.3 RA A.2 + B.2> M8

~~At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable.~~

~~Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests.~~

~~The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent Δk/k.~~

~~During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated Δk/k would limit the magnitude of power excursion resulting from a reduction of moderator density.~~

A2

<Add 3.4.9 RA D.1> M9
<Add SR 3.4.9.1>

<LATER> (3.4A) 3.1.3 Minimum Conditions for Criticality

Specification

3.1.3.1 <LATER> (3.1, 3.4A) The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply. LATER

3.1.3.2 <LATER> (3.4A) Reactor coolant temperature shall be to the right of the criticality limit of Figure 3.1.2-2. LATER

3.1.3.3 <LATER> (3.1, 3.4A) When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization. LATER A1

In MODES 1, 2, 3 + MODE 4 @ ≥ 262°F

3.1.3.4 3.4.9 APPL LCO a ~~The reactor shall be maintained subcritical by at least 1 percent Δk/k~~ until ~~a steam bubble is formed~~ and ~~an indicated~~ water level between 45 and ~~305~~ 320 inches is established in the pressurizer. M9 A9

3.1.3.5 <LATER> (3.1, 3.2) Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. LATER M9

In MODES 1, 2, + 3

3.1.3.6 3.4.9 APPL LCO b RA C.1 RA D.2 ~~The reactor shall not be made critical~~ until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown MODE 4 within the following 12 hours. A3 M9 A1

3.1.3.7 3.4.9 RA A.1/B.1 + <LATER> (3.1 3.4A) With any of the above limits violated, restore the reactor to within the limit in ~~15 minutes~~ 1 hour or be in at least ~~Hot Shutdown~~ MODE 3 within the next ~~15 minutes~~ 6 hours. + LATER L6

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable. A2

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated Δk/k would limit the magnitude of power excursion resulting from a reduction of moderator density.

<Add 3.4.9 RA B.2> M9
<Add 3.4.9 LCO NOTE>

3.1.1
3.1.5

The requirement that the reactor is not to be made critical below the limits of Figure 3.1.2-2 provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than one (1) percent subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a start-up accident and that the water level is above the minimum detectable level.

The requirement that 2 of the 3 emergency-powered pressurizer heaters be operable provides assurance that sufficient heater capacity (≥126 kw) is available to provide reactor coolant system pressure control during a loss of off-site power.

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

A2

REFERENCES

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.5

~~The requirement that the reactor is not to be made critical below the limits of Figure 3.1.2-2 provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.~~

~~If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.~~

~~The requirement for pressurizer bubble formation and specified water level when the reactor is less than one (1) percent subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a start-up accident and that the water level is above the minimum detectable level.~~

~~The requirement that 2 of the 3 emergency powered pressurizer heaters be operable provides assurance that sufficient heater capacity (≥126 kw) is available to provide reactor coolant system pressure control during a loss of off-site power.~~

~~The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.~~

~~The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.~~

~~REFERENCES~~

~~(1) FSAR, Section 3~~

~~(2) FSAR, Section 3.2.2.1.5~~

A2

3.4.2
3.4.3

A2

~~The requirement that the reactor is not to be made critical below the limits of Figure 3.1.2-2 provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.~~

~~If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.~~

~~The requirement for pressurizer bubble formation and specified water level when the reactor is less than one (1) percent subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a start-up accident and that the water level is above the minimum detectable level.~~

~~The requirement that 2 of the 3 emergency-powered pressurizer heaters be operable provides assurance that sufficient heater capacity (≥126 kw) is available to provide reactor coolant system pressure control during a loss of off-site power.~~

~~The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.~~

~~The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.~~

~~<u>REFERENCES</u>~~

~~(1) FSAR, Section 3~~

~~(2) FSAR, Section 3.2.2.1.5~~

A2

The requirement that the reactor is not to be made critical below the limits of Figure 3.1.2-2 provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than one (1) percent subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a start-up accident and that the water level is above the minimum detectable level.

The requirement that 2 of the 3 emergency-powered pressurizer heaters be operable provides assurance that sufficient heater capacity (≥126 kw) is available to provide reactor coolant system pressure control during a loss of off-site power.

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

REFERENCES

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.5

<Insert CTS 23A>

<LATER> (3.4B)

LATER

3.1.4 Reactor Coolant System Activity

Specification

3.1.4.1 Whenever the reactor is operating under steady-state conditions, the following conditions shall be met.

a. The total specific activity of the primary coolant shall not exceed 72/$\bar{E}$ μCi/gm where $\bar{E}$ is the sum of the average beta energy and average gamma energy per disintegration in MEV/disintegration.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

A1

b. The I-131 dose equivalent of the radioiodine activity in the primary coolant shall not exceed 3.5 μCi/gm.

c. If the radioactivity in the primary coolant exceeds the limits given above, corrective action shall be taken immediately to return the coolant activity to within these specifications. If the specific activity limits given above cannot be achieved within 24 hours, the reactor shall be brought to a hot shutdown condition using normal operating procedures. If the coolant radioactivity is not reduced to acceptable limits within an additional 48 hours, the reactor shall be brought to a cold shutdown condition and the cause of the out-of-specification operation ascertained.

<LATER> (3.4B)

LATER

Bases

Rupture of a steam generator tube would allow primary coolant activity to enter the secondary coolant. The major portion of this activity is noble gases and would be released to the atmosphere from the condenser vacuum pump or a relief valve. Activity would continue to be released until the operator could reduce the primary system pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single steam generator tube, followed by isolation of the faulty steam generator within 34 minutes after the tube break. Assuming the full differential pressure across the steam generator, no more than one-quarter of the total primary coolant could be released to the secondary coolant in this period. The decay heat during this period of 1 hour for pressure reduction will generate steam in the secondary system representing less than 15 weight percent of the secondary system.

The parameters assumed in the dose analysis for the single steam generator tube failure included the following values:

1) total primary coolant volume (mass) = 5.2 x 10^5 lbs.

2) total secondary coolant volume (mass) = 2 x 10^6 lbs.

3) leakage rate from primary to secondary system = 1 gpm.

4) fission product decay heat energy for 1 hour = 1.56 x 10^8 BTU.

1.2
1.3
1.4

<INSERT CTS 23A>

< Add Section 1.2, Logical Connectors > A18

< Add Section 1.3, Completion Times > A19

< Add Section 1.4, Frequency > A20

3.1.4 Reactor Coolant System Activity

Specification

MODES 1&2 & MODE 3 w/ RCS T ≥ 500F.

3.4.12 LCO & APPL — 3.1.4.1 ~~Whenever the reactor is operating under steady-state conditions,~~ the following conditions shall be met. — L7

LCO b — a. The total specific activity of the primary coolant shall not exceed 72/E μCi/gm ~~where E is the sum of the average beta energy and average gamma energy per disintegration in MEV/disintegration.~~ — LATER & LA1 Bases

LATER (1.0)

LCO a — b. The I-131 dose equivalent of the radioiodine activity in the primary coolant shall not exceed 3.5 μCi/gm.

3.4.12 RA A.1, RA B.1, RA B.2 — c. If the radioactivity in the primary coolant exceeds the limits given above, ~~corrective action shall be taken immediately to return the coolant activity to within these specifications.~~ If the specific activity limits given above cannot be achieved within 24 hours, the reactor shall be ~~brought to a hot shutdown condition using normal operating procedures.~~ ~~If the coolant radioactivity is not reduced to acceptable limits within an additional 48 hours, the reactor shall be brought to a cold shutdown condition and the cause of the out-of-specification operation ascertained.~~ — L18, M12, L7

in MODE 3 in 6 hours + reduce Tavg <500°F in 12 hours

Bases

A2

~~Rupture of a steam generator tube would allow primary coolant activity to enter the secondary coolant. The major portion of this activity is noble gases and would be released to the atmosphere from the condenser vacuum pump or a relief valve. Activity would continue to be released until the operator could reduce the primary system pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single steam generator tube, followed by isolation of the faulty steam generator within 34 minutes after the tube break. Assuming the full differential pressure across the steam generator, no more than one-quarter of the total primary coolant could be released to the secondary coolant in this period. The decay heat during this period of 1 hour for pressure reduction will generate steam in the secondary system representing less than 15 weight percent of the secondary system.~~

~~The parameters assumed in the dose analysis for the single steam generator tube failure included the following values:~~

1) ~~total primary coolant volume (mass) = 5.2 x 10^5 lbs.~~

2) ~~total secondary coolant volume (mass) = 2 x 10^6 lbs.~~

3) ~~leakage rate from primary to secondary system = 1 gpm.~~

4) ~~fission product decay heat energy for 1 hour = 1.56 x 10^8 BTU.~~

3.4.12

A2

5) steam mass released to environs = 2.84×10^5 lbs.

6) primary coolant released to secondary (34 minutes) = 8.7 x 10^4 lbs.

7) minimum primary to secondary iodine equilibrium activity ratio = 20 to 1 (for 1 gpm leakage).

8) specific I-131 dose equivalent activity = 3.5 μCi/gm (Primary)
= 0.17 μCi/gm (Secondary).

9) gross specific activity in primary = $72/\bar{E}$ μCi/gm.

10) $X/Q = 7.0 \times 10^{-4}$ sec/m³ at limiting point beyond site boundary of 1046 meters for 30 m release height - equivalent to ground level release due to topography including building wake effect for 5 percentile meteorology.

11) total gross radioactivity in primary coolant released to secondary coolant released to environs.

12) ten percent of the combined radioiodine activity from primary activity in secondary coolant and secondary activity present in steam mass (released to environs) assumed released to environs.

The whole body dose resulting from immersion in the cloud containing the released activity would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis employed the simple model of the semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose, because of the short range of beta radiation in air. The resulting whole body dose was determined to be less than 0.5 Rem for this accident.

The thyroid dose from the steam generator tube rupture accident has been analyzed assuming a tube rupture at full load and loss of offsite power at the time of the reactor trip, which results in steam release through the relief valves in the period before the faulty steam generator is isolated and primary system pressure is reduced. The limiting iodine activities for the primary and secondary systems are used in the initial conditions. One-tenth of the iodine contained in the liquid which is converted to steam and passed through the relief valves is assumed to reach the site boundary. The resulting thyroid dose from the combined primary and secondary iodine activity released to the environs was determined to be 1.5 Rem for this accident.

The limit for secondary iodine activity is consistent with the limits on primary system iodine activity and primary-to-secondary leakage of 1 gpm. If the activity should exceed the specified limits following a power transient, the major concern would be whether additional fuel defects had developed bringing the total to above expected levels. From the observed removal of excess activity by decay and cleanup, it should be apparent whether activity is returning to a level below the specification limit. Appropriate action to be taken to bring the activity within specification include one or more of the following: gradual decrease in power to a lower base power, increase in letdown flow rate, and venting of the makeup tank gases to the waste gas decay tanks.

3.4A

3.1.5 Chemistry

Applicability

Applies to the limiting conditions of reactor coolant chemistry for continuous operation of the reactor.

Objective

To protect the reactor coolant system from the effects of impurities in the reactor coolant.

Specification

LA2

TRM

3.1.5.1 The following limits shall not be exceeded for the listed reactor coolant conditions.

Contaminant	Specification	Reactor Coolant Conditions
Oxygen as O_2	0.10 ppm max	above 250°F
Chloride as Cl	0.15 ppm max	above cold shutdown conditions
Fluoride as F	0.15 ppm max	above cold shutdown conditions

L10

3.1.5.2 During operation above 250°F, if any of the specifications in 3.1.5.1 is exceeded, corrective action shall be initiated within 8 hours. If the concentration limit is not restored within 24 hours after initiation of corrective action, the reactor shall be placed in a cold shutdown condition using normal procedures.

3.1.5.3 During operations between 250°F and cold shutdown conditions, if the chloride or fluoride specification in 3.1.5.1 are exceeded, corrective action shall be initiated within 8 hours to restore the normal operating limits. If the specifications are not restored within 24 hours after initiation of corrective action, the reactor shall be placed in a cold shutdown condition using normal procedures.

3.1.5.4 If the oxygen concentration and either the chloride or fluoride concentration of the primary coolant system exceed 1.0 ppm, the reactor shall be immediately brought to the hot shutdown condition using normal shutdown procedures, and action is to be taken immediately to return the system to within normal operation specifications. If specifications given in 3.1.5.1 have not been reached in 12 hours, the reactor shall be brought to a cold shutdown condition using normal procedures.

Bases

A2

By maintaining the chloride, fluoride, and oxygen concentration in the reactor coolant within the specifications, the integrity of the reactor coolant system is protected against potential stress corrosion attack [1,2].

3.4A

Bases - (Continued)

The oxygen concentration in the reactor coolant system is normally expected to be below detectable limits since dissolved hydrogen is used when the reactor is critical and a residual of hydrazine is used when the reactor is subcritical to control the oxygen. The requirement that the oxygen concentration not exceed 0.1 ppm is added assurance that stress corrosion cracking will not occur (3).

If the oxygen, chloride, or fluoride limits are exceeded, measures can be taken to correct the condition (e.g., switch to the spare demineralizer, replace the ion exchanger resin, increase the hydrogen concentration in the makeup tank, etc.) and further because of the time dependent nature of any adverse effects arising from halogen or oxygen concentrations in excess of the limits, it is unnecessary to shutdown immediately.

The oxygen and halogen limits specified are at least an order of magnitude below concentrations which could result in damage to materials found in the reactor coolant system even if maintained for an extended period of time. (3) Thus, the period of eight hours to initiate corrective action and the period of 24 hours thereafter to perform corrective action to restore the concentration within the limits have been established. The eight hour period to initiate corrective action allows time to ascertain that the chemical analyses are correct and to locate the source of contamination. If corrective action has not been effective at the end of 24 hours, then the reactor coolant system will be brought to the cold shutdown condition using normal procedures and corrective action will continue.

The maximum limit of 1 ppm for the oxygen and halogen concentration that will not be exceeded was selected as the hot shutdown limit because these values have been shown to be safe at 500°F. (4)

A2

References

(1) FSAR Section 4.1.2.7
(2) FSAR Section 9.2.2
(3) Corrosion and Wear Handbook, O.J. DePaul, Editor
(4) Stress Corrosion of Metals, Logan

3.4.13
3.4.15

3.1.6 Leakage <Add 3.4.13 Appl.> M13

Specification

3.4.13 LCO c
3.4.13 RA B.1, C.1, C.2

3.1.6.1 If the ~~total~~ identified reactor coolant leakage rate exceeds 10 gpm, the ~~reactor shall be shutdown within 24 hours of detection.~~ restored in 18 hours or in MODE 3 in 6 hours and in MODE 5 in 36 hours L14 M13

3.4.13 LCO b
3.4.13 RA B.1, C.1, C.2

3.1.6.2 If unidentified reactor coolant leakage ~~(exceeding normal evaporative losses)~~ exceeds 1 gpm ~~or if any reactor coolant leakage is evaluated as unsafe~~ the reactor shall be ~~shutdown within 24 hours of detection.~~ restored in 18 hours or in MODE 3 in 6 hours and in MODE 5 in 36 hours A11 LA2 TRM M13

3.4.13 LCO a
3.4.13 RA C.1, C.2

3.1.6.3.a If it is determined that any ~~reactor coolant~~ pressure boundary leakage exists ~~through a non-isolable fault in a reactor coolant system strength boundary (such as the reactor vessel, piping, valve body, etc., except steam generator tubes)~~, the reactor shall ~~be shutdown and a cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.~~ in MODE 3 in 6 hours and in MODE 5 in 36 hours. LA1 BASES M13

3.4.13 LCO d
3.4.13 RA A.1, C.1, C.2

3.1.6.3.b If the leakage through the tubes of any one steam generator equals or exceeds 150 gallons per day ~~(0.104 gpm)~~, ~~a reactor shutdown shall be initiated within 4 hours and the reactor shall be in the cold shutdown condition within the next 30 hours.~~ Restore in 4 hours or be in MODE 3 in 6 hours and in MODE 5 in 36 hours LA1 BASES L3 M13

3.1.6.4 ~~Deleted~~ A1

3.1.6.5 ~~Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude of the leak, shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10CFR20.~~ LA2 TRM

3.1.6.6 ~~If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, or 3.1.6.3 the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.~~ A4

3.4.15 LCO
3.4.15 Appl

3.4.15 RA B.2
3.4.15 RA B.1.1
3.4.15 RA C.1
3.4.15 RA C.2

3.1.6.7 In MODES 1, 2, 3 & 4, ~~When the reactor is at power operation~~ ~~three~~ reactor coolant leak detection systems of different operating principles shall be in operation. One of these systems is sensitive to radioactivity and consists of a radioactive gas detector ~~and~~ or an air particulate activity detector. Both of these instruments may be out-of-service simultaneously for a period of no more than ~~72 hours~~ provided ~~two~~ other means are available to detect leakage and reactor building air samples are taken and analyzed in the laboratory at least once ~~per shift~~; otherwise, be in at least ~~Hot Standby~~ MODE 3 within the next 6 hours and in ~~Cold Shutdown~~ MODE 5 within ~~the following~~ 30 hours. M13 L13 L13 M6

See page 27-2

3.1.6.8 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which

<Add 3.4.15 Cond. A & RA B.1.2 with Note> L13

<Add 3.4.15 Actions Note> L13

<Add 3.4.15 Cond. D> M15

3.1.6 Leakage

Specification

3.1.6.1 ~~If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shutdown within 24 hours of detection.~~ L19

See page 27-1

3.1.6.2 If unidentified reactor coolant leakage (exceeding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shutdown within 24 hours of detection.

3.1.6.3.a If it is determined that any reactor coolant leakage exists through a non-isolable fault in a reactor coolant system strength boundary (such as the reactor vessel, piping, valve body, etc., except steam generator tubes), the reactor shall be shutdown and a cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.3.b If the leakage through the tubes of any one steam generator equals or exceeds 150 gallons per day (0.104 gpm), a reactor shutdown shall be initiated within 4 hours and the reactor shall be in the cold shutdown condition within the next 30 hours.

3.1.6.4 Deleted

3.1.6.5 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude of the leak, shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10CFR20.

See page 27-1

3.1.6.6 ~~If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, or 3.1.6.3 the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.~~ L19

See page 27-1

3.1.6.7 When the reactor is at power operation, three reactor coolant leak detection systems of different operating principles shall be in operation. One of these systems is sensitive to radioactivity and consists of a radioactive gas detector and an air particulate activity detector. Both of these instruments may be out-of-service simultaneously for a period of no more than 72 hours provided two other means are available to detect leakage and reactor building air samples are taken and analyzed in the laboratory at least once per shift; otherwise, be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.1.6.8 ~~Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which~~ LA2 TRM

M5

3.4.13
3.4.14
3.4.15

<Add 3.4.14 Appl> A7
<Add 3.4.14 ACTIONS Note 1> A7

~~vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1 and 3.1.6.6 except that such losses when added to leakage shall not exceed 30 gpm.~~ M5 LA2 TRM L19

3.1.6.9 If the reactor coolant system pressure isolation valve leakage is greater than the values given in Table 3.1.6.9, isolate (by having at least two valves in the high pressure piping closed*) the high pressure portion of the affected system from the low pressure portion within 4 hours and ~~apply Specification 3.3.6~~, or be in at least ~~hot shutdown~~ MODE 3 within the next 6 hours and in ~~cold shutdown~~ MODE 5 within the following 30 hours. A1

3.4.14
RA A.1
ACTIONS Note 2
RA C.1/C.2

Bases

~~Every reasonable effort will be made to reduce reactor coolant leakage, including evaporative losses (which may be on the order of 0.5 gpm), to prevent a large leak from masking the presence of a smaller leak. Reactor building sump level, water inventory balances, radiation monitoring equipment, 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 radioactive contamination and cleanup or it could develop into a still more serious problem; and therefore, the first indication of such leakage will be followed up as soon as practicable.~~ A2

~~Although some leak rates on the order of GPM may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks on the order of drops per minute 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, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.~~

~~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 Operating Staff and will be documented in writing and approved by the Superintendent. Under these conditions, an allowable reactor coolant system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also available even during a loss of off-site power.~~

~~If leakage is to the reactor building it may be identified by one or more of the following methods:~~

~~a. Leakage is monitored by a level indicator in the reactor building sump. Changes in normal sump level may be indicative of leakage from any of the systems located inside the reactor building such as the reactor coolant system, service water system, intermediate cooling system and steam and feedwater lines or condensation of humidity within the reactor building atmosphere. The reactor building sump contains 63.6 gallons per inch of height. A 1 gpm leak would be detected in less than 1 hour.~~

3.4.14 RA A.1 *The motor operated valve shall remain closed and ~~power supplies deenergized~~ deactivated. A1

3.4.13
3.4.14
3.4.15

A2

b. Total reactor coolant system leakage rate is periodically determined by comparing indications of reactor power, reactor coolant temperature, pressurizer water level and reactor coolant makeup tank level over a time interval. All of these indications are recorded. Since the pressurizer level is maintained essentially constant by the pressurizer level controller, any coolant leakage is replaced by coolant from the reactor coolant makeup tank resulting in a tank level decrease. The reactor coolant makeup tank capacity is 31 gallons per inch of height and each graduation on the level recorder represents 2 inches of tank height. This inventory monitoring method is capable of detecting changes on the order of 62 gallons. A 1 gpm leak would therefore be detectable within approximately 1.1 hours.

As described above, in addition to direct observation, the means of detecting reactor coolant leakage are based on different principles, i.e., activity, sump level and reactor coolant inventory measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the reactor building.

c. The reactor building gaseous monitor is sensitive to low leak rates if expected values of failed fuel exist. The rates of reactor coolant leakage to which the instrument is sensitive are discussed in FSAR Section 4.2.3.8.

The upper limit of 30 gpm is based on the contingency of a hypothetical loss of all AC power. A 30 gpm loss of water in conjunction with a hypothetical loss of all AC power and subsequent cooldown of the reactor coolant system by the atmospheric dump system and steam driven emergency feedwater pump would require more than 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. This will be ample time to restore both electrical power to the station and makeup flow to the reactor coolant system.

The steam generator tube leakage limit (i.e., primary to secondary leakage limit) in Specification 3.1.6.3 is intended to assure timely shutdown of the plant for appropriate corrective action before rupture of the steam generator tubes occurs under normal operating or postulated accident conditions. These limits also serve to provide added assurance that the dosage contribution from tube leakage will be limited to a small fraction of 10CFR100 limits for a design basis steam generator tube rupture or main steam line break event.

References

FSAR Section 4.2.3.8

(Add SR 3.4.14.1, Note) L16

LA1 Bases

SR 3.4.14.1

TABLE 3.1.6.8

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum (a)(b)(c) Allowable Leakage
Decay Heat Removal Train A	DH-14A	≤ 5.0 GPM
	DH-13A) AND) DH-17)	≤ 5.0 GPM (both valves together total)
Decay Heat Removal Train B	DH-14B	≤ 5.0 GPM
	DH-13B) AND) DH-18)	≤ 5.0 GPM (both valves together total)

A1

Footnote:

(a) 1. ~~Leakage rates less than or equal to 1.0 gpm are considered acceptable.~~

2. ~~Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.~~

3. ~~Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.~~

L5

SR 3.4.14.1 4. Leakage rates greater than 5.0 gpm are considered unacceptable.

SR 3.4.14.1 (b) Minimum differential test pressure shall not be less than 150 psig.

(c) ~~To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.~~

LA1 Bases

3.1.1
3.1.3
3.1.9

MTC — A1

3.1.7 Moderator Temperature Coefficient of Reactivity Specification

3.1.3 LCO — 3.1.7.1 The moderator temperature coefficient (MTC) shall be non-positive whenever thermal power is ≥95% of rated thermal power and shall be less positive than 0.9 x 10^{-4} Δk/k/°F whenever thermal power is <95% of rated thermal power ~~and the reactor is not shutdown~~.

3.1.3 Appl. — MODES 1 and 2 — M4

SR 3.1.3.1 3.1.7.2 The MTC shall be determined to be within its limits by in MODE 1 confirmatory measurements prior to initial operation ~~above 5% of rated thermal power~~ after each fuel loading. ~~MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the limits in 3.1.7.1 above.~~ — A1 — LA1 BASES

3.1.3 RA A.1 3.1.7.3 With the MTC outside any one of the above limits, be in at least ~~HOT STANDBY~~ MODE 3 within 6 hours. — M12

~~Bases~~

~~A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power, the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of +0.9 x 10^{-4} Δk/k/°F corrected to 95% of rated power. The most limiting event for positive MTC, the Startup Accident, has been analyzed for a range of moderator temperature coefficients including +0.9 x 10^{-4} Δk/k/°F.~~ — A2

<Add 3.1.9 PHYSICS TESTS exception to LCO 3.1.3> — L9

<Add 3.1.1 SHUTDOWN MARGIN (SDM)> — M6

3.1.8
3.1.9

3.1.8 ~~Low Power~~ Physics ~~Testing Restrictions~~ Exceptions – MODE 2 (A1)

~~Specification~~

~~The following special limitations are placed on low power physics testing.~~ (A1)

3.1.8.1 Reactor Protective System Requirements (A8)

3.1.9.a LCO — A. Below 1720 psig, shutdown bypass trip setting limits shall apply in accordance with Table 2.3-1.

3.1.9.b LCO — B. Above 1800 psig, nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1.

3.1.9.c LCO — 3.1.8.2 Startup rate rod withdrawal hold [1] shall be in effect at all times.

3.1.8.3 ~~During low power physics testing the minimum reactor coolant temperature for criticality shall be to the right of the criticality limit of Figure 3.1.2-2.~~ (A7)

3.1.8.d LCO
3.1.9.d LCO — The shutdown margin shall be maintained greater than or equal to that specified in the COLR with the highest worth control rod fully withdrawn.

~~Bases~~

~~The above specification provides additional safety margins during low power physics testing.~~

~~REFERENCES~~

~~(1) FSAR, Section 7.2.2.1.3.~~ (A2)

<ADD 3.1.8 LCO a, b, c with Note; Appl.; Actions SR 3.1.8.1; SR 3.1.8.2 with Note; SR 3.1.8.3; SR 3.1.8.4> (M14)

<Add 3.1.9 Appl> (A8)

<Add 3.1.9 ACTIONS and SRs> (M14)

<Add LCO 3.1.8 & LCO 3.1.9 PHYSICS TESTS exceptions to LCO 3.1.4 and LCO 3.1.6.> (L10)

TSI-313-07-00 A1

3.1.9 Control Rod Operation

LA1 TRM

Specification

3.1.9.1 The concentration of dissolved gases in the reactor coolant shall be limited to 100 std. cc/liter of water at the reactor vessel outlet temperature.

3.1.9.2 Allowable combinations of pressure and temperature for control rod operation shall be to the left of and above the limiting pressure versus temperature curve for a dissolved gas concentration of 100 std. cc/liter of water as shown in Figure 3.1.9-1.

3.1.9.3 In the event the limits of Specifications 3.1.9.1 or 3.1.9.2 are exceeded, the vessel level instrument vent shall be checked for accumulation of undissolved gases. The temperature, pressure and dissolved gas concentration shall be restored to within their limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

L14

Bases

A2

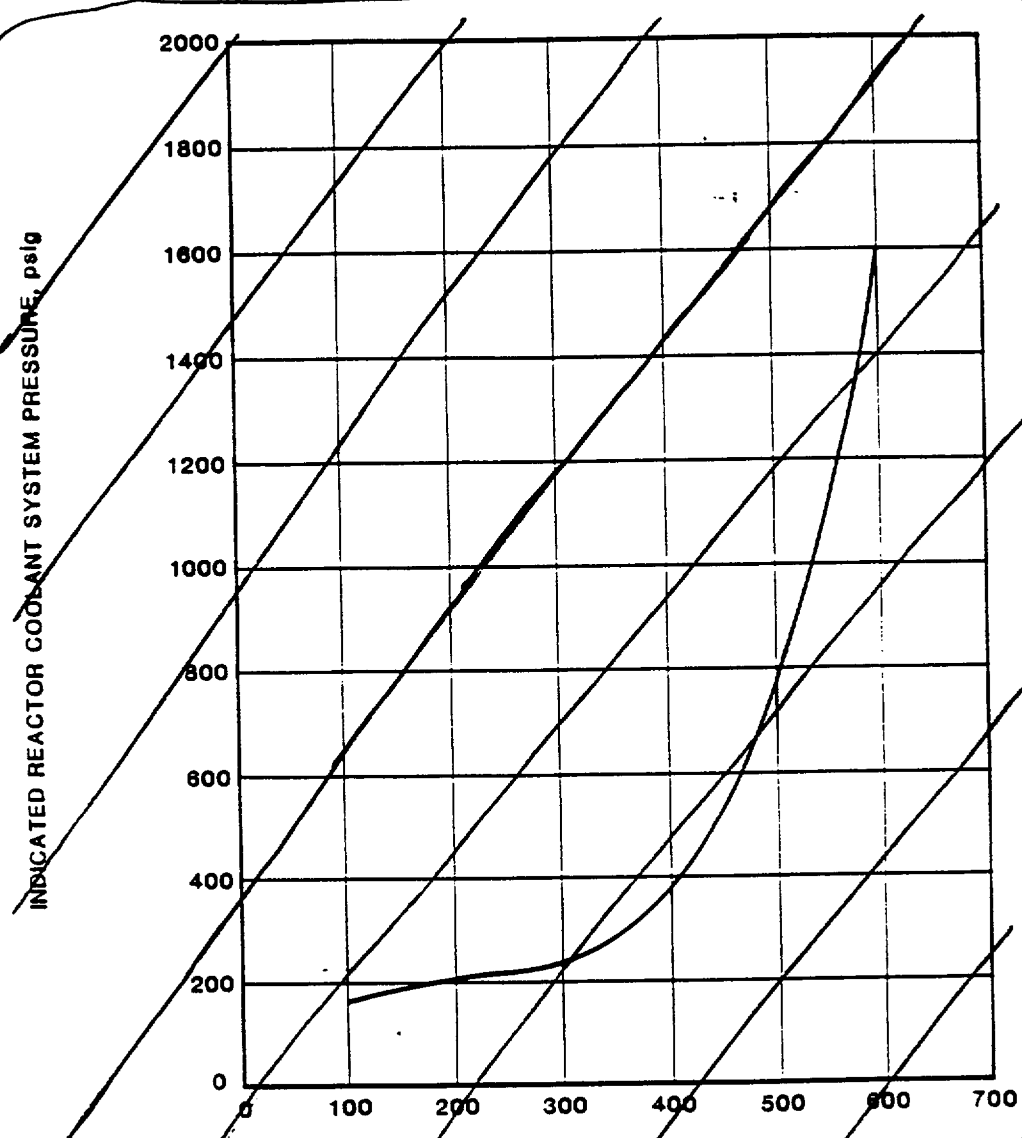
By maintaining the reactor coolant temperature and pressure as specified above, any dissolved gases in the reactor coolant system are maintained in solution.

Although the dissolved gas concentration is expected to be approximately 20-40 std. cc/liter of water, the dissolved gas concentration is conservatively assumed to be 100 std. cc/liter of water at the reactor vessel outlet temperature.

The limiting pressure versus temperature curve for dissolved gases is determined by the equilibrium pressure versus temperature curve for the dissolved gas concentration of 100 std. cc/liter of water. The equilibrium total pressure is the sum of the partial pressure of the dissolved gases plus the partial pressure of water at a given temperature. The margin of error consists of the maximum pressure difference between the pressure sensing tap and lowest pressure point in the system, the maximum pressure gage error, and the pressure difference due to the maximum temperature gage error.

If either the maximum dissolved gas concentration (100 std. cc/liter of water) is exceeded or the operating pressure falls below the limiting pressure versus temperature curve, the vessel level instrument vent should be checked for accumulation of undissolved gases.

3.1



LA1
TRM

Limiting Pressure VS Temperature for Control Rod Drive Operation With 100 STD CC/Liter H_2O

Figure 3.1.9-1

3.5

3.2 MAKEUP AND CHEMICAL ADDITION SYSTEMS

A1

Applicability

Applies to the operational status of the makeup and the chemical addition systems.

Objective

To provide for adequate boration under all operating conditions to assure ability to bring the reactor to a cold shutdown condition.

LA3

TRM

Specification

3.2.1 The reactor shall not be heated or maintained above 200°F unless the following conditions are met:

3.2.1.1 Two makeup pumps are operable except as specified in Specification 3.3.

3.2.1.2 A source of concentrated boric acid solution in addition to that in the borated water storage tank is available and operable. This requirement is fulfilled by the boric acid addition tank and one associated boric acid pump being operable. This tank shall contain at least the equivalent of the boric acid volume and concentration requirements of Figure 3.2-1 as boric acid solution with a temperature of at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the makeup system shall also be operable and shall have a temperature of at least 10°F above the crystallization temperature for the concentration in the tank.

L11

3.2.1.3 The boric acid addition tank and associated piping, valves and both pumps may be out of service for a maximum of 24 hours. After the 24 hour period, if the system is not returned to service and operable, the reactor shall be brought to the hot shutdown condition within an additional 12 hours.

A2

Bases

The makeup system and chemical addition system provide control of the reactor coolant system boron concentration. (1) This is normally accomplished by using any of the three makeup pumps in series with a boric acid pump associated with the boric acid addition tank. The alternate method of boration will be the use of the makeup pumps taking suction directly from the borated water storage tank. (2)

The quantity of boric acid in storage from either of the two above mentioned sources is sufficient to borate the reactor coolant system to a 1% subcritical margin in the cold condition (200°F) at the worst time in core life with a stuck control rod assembly and after xenon decay.

3.5.

Minimum volumes (including a 10% safety factor) as specified by Figure 3.2-1 for the boric acid addition tank or an operable borated water storage tank (3) will each satisfy this requirement. The specification assures that adequate supplies are available whenever the reactor is heated above 200°F so that a single failure will not prevent boration to a cold condition. The minimum volumes of boric acid solution given include the boron necessary to account for xenon decay.

The principal method of adding boron to the primary system is to pump the concentrated boric acid solution (8700 ppm boron, minimum) into the makeup tank using the 25 gpm boric acid pumps.

The alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps.

Concentration of boron in the boric acid addition tank may be higher than the concentration which would crystallize at ambient conditions. For this reason and to assure a flow of boric acid is available when needed this tank and its associated piping will be kept 10°F above the crystallization temperature for the concentration present. Once in the makeup system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

A2

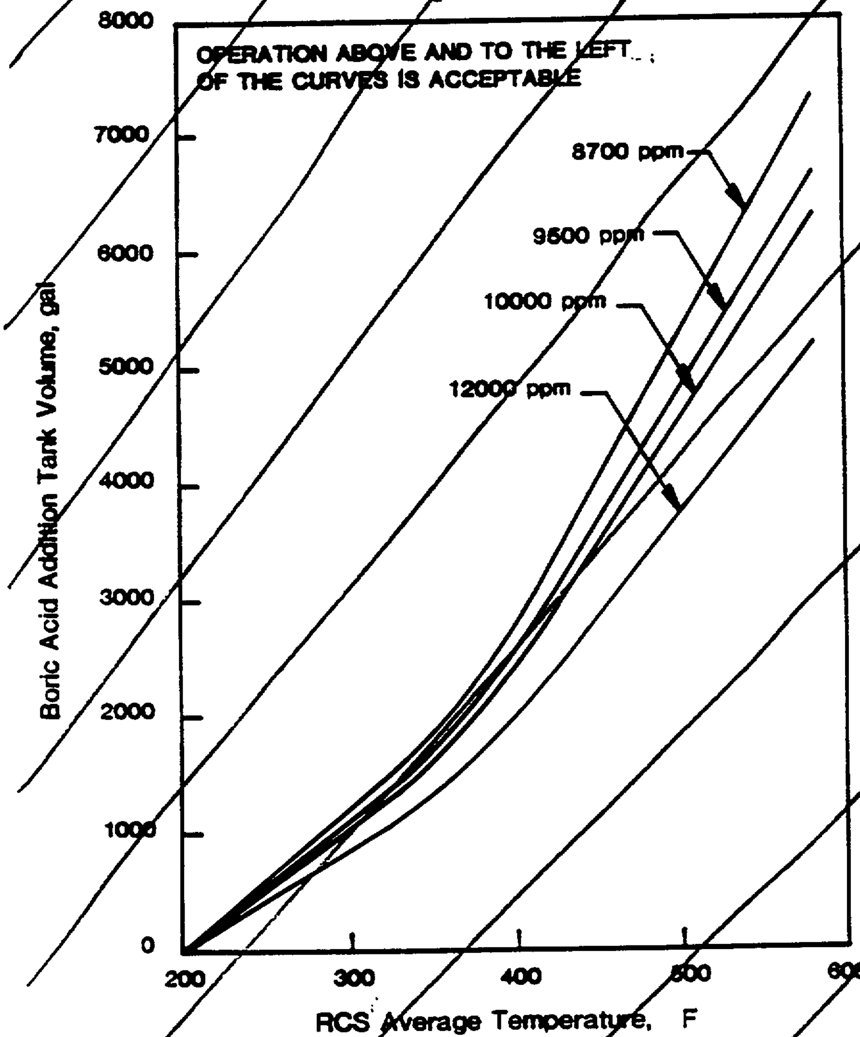
REFERENCES

1. FSAR, Section 9.1; 9.2
2. FSAR, Figure 6-2
3. SAR, Section 3.1

3.5

Boric Acid Addition Tank Volume and Concentration Vs RCS Average Temperature - ANO-1, Cycle 8

Figure 3.2-1



Temp., °F	Required Volume, gal			
	8700 ppm	9500 ppm	10000 ppm	12000 ppm
579	7308	6657	6306	5200
532	6126	5580	5289	4355
500	5273	4802	4548	3749
400	2790	2543	2409	1984
300	1234	1129	1070	877
200	0	0	0	0

LA3 TRM

3.3.15

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTO BUILDING SPRAY SYSTEMS

~~**Applicability**~~

~~Applies to the emergency core cooling, reactor building emergency cooling and reactor building spray systems.~~

~~**Objectivity**~~

~~To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor buildi spray systems.~~

A1

Specification

in MODES 1, 2, & 3;

M6

& LATER

3.3.1 The following equipment shall be operable ~~whenever containment integrity is established as required by Specification 3.6.1:~~

L3

3.3.15 #15 Appl. & ⟨LATER⟩ (3.5, 3.6, 3.7)

(A) ~~One reactor building spray pump and its associated spray noz header.~~

⟨LATER⟩ (3.6)

(B) ~~One train of reactor building emergency cooling.~~

LATER

⟨LATER⟩ (3.7)

(C) ~~Two out of three service water pumps shall be operable, powe from independent essential buses, to provide redundant and independent flow paths.~~

LATER

⟨LATER⟩ (3.5)

(D) ~~Two engineered safety feature actuated Low Pressure Injectio (LPI) pumps shall be operable.~~

LATER

⟨LATER⟩ (3.7)

(E) ~~Both low pressure injection coolers and their cooling water supplies shall be operable.~~

LATER

Table 3.3.15-1, #15 (F) Two Borated Water Storage Tank (BWST) level instrument channels shall be operable.

⟨LATER⟩ (3.5)

(G) ~~The borated water storage tank shall contain a level of 40.2 ± 1.8 ft. (387,400 ± 17,300 gallons) of water having a concentration of 2470 ± 200 ppm boron at a temperature not less than 40F. The manual valve on the discharge line from the borated water storage tank shall be locked open.~~

LATER

(H) ~~The four reactor building emergency sump isolation valves to the LPI system shall be either manually or remote-manually operable.~~

3.5.2
3.5.3
3.5.4

3.3 EMERGENCY CORE COOLING ~~REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS~~ (ECCS) — A1

~~Applicability~~

~~Applies to the emergency core cooling, reactor building emergency cooling and reactor building spray systems.~~

~~Objectivity~~

~~To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor building spray systems.~~

~~Specification~~ — A1

3.5.2 Appl. (for LPI)
3.5.3 Appl.
3.5.4 Appl.
& <LATER>(3.3D, 3.6, 3.7)

3.3.1 The following equipment shall be operable MODES 1, 2, 3, 4 ~~whenever containment integrity is established as required by Specification 3.6.1:~~ — M8 & LATER

<LATER> (3.6)

~~(A) One reactor building spray pump and its associated spray nozzle header.~~

~~(B) One train of reactor building emergency cooling.~~ — LATER

<LATER> (3.7)

~~(C) Two out of three service water pumps shall be operable, power from independent essential buses, to provide redundant and independent flow paths.~~ — LATER

3.5.2 LCO
3.5.3 LCO

(D) Two engineered safety feature actuated Low Pressure Injection (LPI) pumps shall be operable.

<LATER> (3.7)

~~(E) Both low pressure injection coolers and their cooling water supplies shall be operable.~~ — LATER

<LATER> (3.3D)

~~(F) Two Borated Water Storage Tank (BWST) level instrument channel shall be operable.~~ — LATER

LA1 BASES

3.5.4 LCO

(G) The borated water storage tank shall contain a level of 40.2 ± 1.8 ft. ~~(387,400 ± 17,300 gallons)~~ of water having a concentration of 2470 ± 200 ppm boron at a temperature not less than 40F. — A9 ~~The manual valve on the discharge line from the borated water storage tank shall be locked open.~~ — LA3 SAR

3.5.2 LCO
3.5.3 LCO

~~(H) The four reactor building emergency sump isolation valves to the LPI system shall be either manually or remote-manually operable.~~ — LA1 BASES

<Add SR 3.5.4.1 & Note> — M7

<Add SR 3.5.4.2> — M7

5

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTC BUILDING SPRAY SYSTEMS

~~Applicability~~

~~Applies to the emergency core cooling, reactor building emergency cooling and reactor building spray systems.~~ A1

~~Objectivity~~

~~To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor buildi spray systems.~~

Specification

In MODES 1, 2, 3 & 4

3.6.5 LCO
3.6.5 Appl.
& <Later> (3.3D, 3.5, 3.7)

3.3.1 The following equipment shall be operable ~~whenever containment integrity is established as required by Specification 3.6.1:~~ M1 & Later

3.6.5 LCO Note

(A) One reactor building spray pump and its associated spray nozzle header.

3.6.5 LCO Note

(B) One train of reactor building emergency cooling.

<Later> (3.7)

~~(C) Two out of three service water pumps shall be operable, powe from independent essential buses, to provide redundant and independent flow paths.~~ Later

<Later> (3.5)

~~(D) Two engineered safety feature actuated Low Pressure Injectic (LPI) pumps shall be operable.~~ Later

<Later> (3.7)

~~(E) Both low pressure injection coolers and their cooling water supplies shall be operable.~~ Later

<Later> (3.3D)

~~(F) Two Borated Water Storage Tank (BWST) level instrument chanr shall be operable.~~ Later

<Later> (3.5)

~~(G) The borated water storage tank shall contain a level of 40.2 ± 1.8 ft. (387,400 ± 17,300 gallons) of water having a concentration of 2470 ± 200 ppm boron at a temperature not less than 40F. The manual valve on the discharge line from the borated water storage tank shall be locked open.~~ Later

~~(H) The four reactor building emergency sump isolation valves to the LPI system shall be either manually or remote-manually operable.~~

3.7.7

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS

Applicability

Applies to the emergency core cooling, reactor building emergency cooling and reactor building spray systems.

A1

Objectivity

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor building spray systems.

Specification

MODES 1, 2, 3 & 4

3.7.7 Appl. & <LATER> (3.3D, 3.5, 3.6)

3.3.1 The following equipment shall be operable whenever containment integrity is established as required by Specification 3.6.1: M9

<LATER> (3.6)

(A) One reactor building spray pump and its associated spray nozzle header. — LATER

(B) One train of reactor building emergency cooling loops A1

3.7.7 LCO

(C) Two out of three service water pumps shall be operable, powered from independent essential buses, to provide redundant and independent flow paths. LA3 SAR

<LATER> (3.5)

(D) Two engineered safety feature actuated Low Pressure Injection (LPI) pumps shall be operable. — LATER

3.7.7 LCO

(E) Both low pressure injection coolers and their cooling water supplies shall be operable.

<LATER> (3.3D)

(F) Two Borated Water Storage Tank (BWST) level instrument channel shall be operable. — LATER

<LATER> (3.5)

(G) The borated water storage tank shall contain a level of 40.2 ± 1.8 ft. (387,400 ± 17,300 gallons) of water having a concentration of 2470 ± 200 ppm boron at a temperature not less than 40F. The manual valve on the discharge line from the borated water storage tank shall be locked open. — LATER

(H) The four reactor building emergency sump isolation valves to the LPI system shall be either manually or remote-manually operable.

3.5.1
3.5.2
3.5.3
3.5.4

3.5.2 (for LPI), 3.5.3, 3.5.4 LCO

& ⟨LATER⟩ (3.6, 3.7)

Sealed, or otherwise secured

L10 & LATER

(I) The engineered safety features valves associated with each of the above systems shall be operable or locked in the ES position.

A1

3.3.2 3.5.2 Appl. (for HPI) ~~In addition to 3.3.1 above,~~ the following ECCS equipment shall be operable when the reactor coolant system is above 350F ~~and irradiated fuel is in the core:~~ A4 A1

3.5.2 LCO

(A) Two ~~out of three~~ high pressure injection (makeup) pumps shall be ~~maintained~~ operable, ~~powered from independent essential buses, to provide redundant and independent flow paths.~~ LA3 SAR

(B) Engineered safety features valves associated with 3.3.2.a above shall be operable or locked in the ES position. Sealed, or otherwise Secured L10 A1

3.3.3 3.5.1 Appl. ~~In addition to 3.3.1 and 3.3.2 above, the following ECCS equipment shall be operable when the reactor coolant system is above 800 psig:~~

3.5.1 LCO

(A) The two core flooding tanks shall each contain an indicated minimum of ~~13 ± 0.4 feet~~ (1040 ± ~~30~~ 70 ft³) of borated water at 600 ± ~~25~~ 40 psig. A12 A8

(B) Core flooding tank boron concentration shall not be less than 2270 ppm boron.

(C) The electrically operated discharge valves from the core flood tanks shall be open ~~and breakers locked open and tagged.~~ LA3 SAR

~~(D) One of the two pressure instrument channels and one of the two level instrument channels per core flood tank shall be operable.~~ LA3 TRM

~~3.3.4 The reactor shall not be made critical unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 above is operable.~~ LATER

⟨LATER⟩ (3.6)

~~(A) Two reactor building spray pumps and their associated spray nozzle headers and two trains of reactor building emergency cooling. The two reactor building spray pumps shall be powered from operable independent emergency buses and the two reactor building emergency cooling trains shall be powered from operable independent emergency buses.~~

~~(B) The sodium hydroxide tank shall contain a volume of ≥29,000 gallons of sodium hydroxide solution at a concentration >5.0 wt% and <16.5 wt%.~~

~~(C) All manual valves in the main discharge lines of the sodium hydroxide tanks shall be locked open.~~

⟨Add SR 3.5.1.5⟩ M5

⟨Add SR 3.5.1.1, SR 3.5.1.2, & SR 3.5.1.3⟩ M4

A13

3.6.5
3.6.6

3.6.5 LCO & <LATER> (3.5, 3.7) — (I) The engineered safety features valves associated with each of the above systems shall be operable or locked in the ES position. — sealed, or otherwise secured — L19 & LATER

3.3.2 In addition to 3.3.1 above, the following ECCS equipment shall be operable when the reactor coolant system is above 350F and irradiated fuel is in the core: — <LATER> (3.5) — LATER

(A) Two out of three high pressure injection (makeup) pumps shall be maintained operable, powered from independent essential buses, to provide redundant and independent flow paths.

(B) Engineered safety features valves associated with 3.3.2.a above shall be operable or locked in the ES position.

3.3.3 In addition to 3.3.1 and 3.3.2 above, the following ECCS equipment shall be operable when the reactor coolant system is above 800 psig:

(A) The two core flooding tanks shall each contain an indicated minimum of 13 ± 0.4 feet (1040 ± 30 ft³) of borated water at 600 ± 25 psig.

(B) Core flooding tank boron concentration shall not be less than 2270 ppm boron.

(C) The electrically operated discharge valves from the core flood tanks shall be open and breakers locked open and tagged.

(D) One of the two pressure instrument channels and one of the two level instrument channels per core flood tank shall be operable.

in MODE 1 or 2 — A1

3.3.4 The reactor shall not be ~~made critical~~ unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 above is operable.

3.6.5 LCO & Appl.
3.6.6 LCO & Appl.

(A) Two reactor building spray pumps and their associated spray nozzle headers and two trains of reactor building emergency cooling. ~~The two reactor building spray pumps shall be powered from operable independent emergency buses and the two reactor building emergency cooling trains shall be powered from operable independent emergency buses.~~ — LA1 Bases — M7

SR 3.6.6.2
SR 3.6.6.3

(B) verify once each 184 days that The sodium hydroxide tank ~~shall~~ contains a volume of ≥9,000 gallons of sodium hydroxide solution at a concentration >5.0 wt% and <16.5 wt%.

3.6.6 LCO (C) All manual valves in the main discharge lines of the sodium hydroxide tanks shall be locked open. — L14

<Add SR 3.6.6.1 & SR 3.6.6.4> — M10

LAR

A21

3.7.7

⟨Add SR 3.7.7.1 with Note⟩ — M10

L19

3.7.7 LCO & ⟨LATER⟩ (3.5, 3.6)

(I) The engineered safety features valves associated with each of the above systems shall be operable or locked in the ES position. *Sealed, or otherwise secured* — *C & LATER*

3.3.2 In addition to 3.3.1 above, the following ECCS equipment shall be operable when the reactor coolant system is above 350F and irradiated fuel is in the core: — ⟨LATER⟩ (3.5) — LATER

(A) Two out of three high pressure injection (makeup) pumps shall be maintained operable, powered from independent essential buses, to provide redundant and independent flow paths.

(B) Engineered safety features valves associated with 3.3.2a above shall be operable or locked in the ES position.

3.3.3 In addition to 3.3.1 and 3.3.2 above, the following ECCS equipment shall be operable when the reactor coolant system is above 800 psig:

(A) The two core flooding tanks shall each contain an indicated minimum of 13 ± 0.4 feet (1040 ± 30 ft^3) of borated water at 600 ± 25 psig.

(B) Core flooding tank boron concentration shall not be less than 2270 ppm boron.

(C) The electrically operated discharge valves from the core flood tanks shall be open and breakers locked open and tagged.

(D) One of the two pressure instrument channels and one of the two level instrument channels per core flood tank shall be operable.

3.3.4 The reactor shall not be made critical unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 above is operable. — ⟨LATER⟩ (3.6) — LATER

(A) Two reactor building spray pumps and their associated spray nozzle headers and two trains of reactor building emergency cooling. The two reactor building spray pumps shall be powered from operable independent emergency buses and the two reactor building emergency cooling trains shall be powered from operable independent emergency buses.

(B) The sodium hydroxide tank shall contain a volume of ≥9,000 gallons of sodium hydroxide solution at a concentration >5.0 wt% and <16.5 wt%.

(C) All manual valves in the main discharge lines of the sodium hydroxide tanks shall be locked open.

A12

3.5.1
3.5.2
3.5.3
3.5.4

3.5.1, 3.5.2, 3.5.3, 3.5.4 LCO

& ⟨LATER⟩ (3.6, 3.7)

(D) Engineered safety feature valves and interlocks associated with 3.3.1, 3.3.2, and 3.3.3 shall be operable or locked, sealed, or otherwise secured in the ES position. L10 & LATER

~~3.3.5 Maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, service water reactor building spray and reactor building emergency cooling~~ L4 LATER

⟨LATER⟩ (3.6, 3.7)

3.6.5
3.6.6

3.6.5 LCO
3.6.6 LCO
&
⟨Later⟩
(3.5 3.7)

(D) Engineered safety feature valves and interlocks associated with 3.3.1, 3.3.2, and 3.3.3 shall be operable or locked, ⟨sealed, or otherwise secured⟩ in the ES position. L19 & LATER

⟨Later⟩
(3.5, 3.7)

3.3.5 ~~Maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, service water reactor building spray and reactor building emergency cooling~~ L3 Later

3.7.7 LCO

<LATER> (3.5, 3.6)

(D) Engineered safety feature valves and interlocks associated with 3.3.1, 3.3.2, and 3.3.3 shall be operable or locked, sealed, or otherwise secured in the ES position.

L19

LATER

3.3.5 ~~Maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, service water reactor building spray and reactor building emergency cooling~~

<LATER> (3.5, 3.6)

L13

LATER

<LATER> (3.5, 3.6, 3.7) systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance. — LATER

3.3.15 PAM #15 RA F.1/F.2 & <LATER> (3.5, 3.6, 3.7)

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition within 36 hours, and, if not corrected, in cold shutdown condition within an additional 72 hours.

6 — MODE 3 — 12 — MODE 4 — M10 + LATER — A1

3.3.7 Exceptions to 3.3.6 shall be as follows: — M10

3.3.15 PAM #15 RA A.1

(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.

30 — L11 — A1

<LATER> (3.5)

(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CNT instrument channel (pressure of level) shall be operable. — LATER

<LATER> (3.6)

(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within 7 days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. — LATER

(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

<Add 3.3.15 RA B.1 & RA C.1 for PAM #15> — L11

3.7.7

<Add 3.7.7 RA A.1 Notes 1 & 2> A4

3.7.7 RA A.1 & <LATER> (3.5, 3.6)

~~systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance.~~ L13 & LATER

& <LATER> (3.3D, 3.5, 3.6) 3.7.7 RA A.1 3.7.7 RA B.1 3.7.7 RA B.2

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated ~~and the reactor shall be in hot shutdown~~ MODE 3 condition within 36 hours, and, if not corrected, in ~~cold shutdown~~ MODE 5 condition within an additional 72 hours. L13 & LATER A1 M11

~~3.3.7 Exceptions to 3.3.6 shall be as follows:~~

<LATER> (3.3D)

~~(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.~~ LATER

<LATER> (3.5)

~~(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure level) shall be operable.~~ LATER

<LATER> (3.6)

~~(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.~~ LATER

~~(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 7 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.~~

<Add 3.5.1 Condition A> — L1

<Add 3.5.1 Condition C - Second entry condition> — A3

See page 38-2 & 38-3

<LATER> (3.6)

systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which woul make the affected system train inoperable for more than 24 consecutiv hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstra to be operable within 24 hours prior to the maintenance. — Later

See page 38-2

3.5.1 Cond. B

3.5.1 Cond. C

& <Later> (3.3D, 3.6, 3.7)

3.3.6 If the conditions of Specifications ~~3.3.1, 3.3.2~~, 3.3.3, ~~3.3.4 and 3.3.5~~ cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated ~~and~~ the reactor shall be in ~~hot shutdown condition~~ MODE 3 within ~~36~~ 6 hours, and, if not corrected, in ~~cold shutdown condition~~ MODE 3 with RCS pressure ≤ 800 psig within ~~an additional 72~~ 12 hours. within 6 hours OR — M2 & LATER — L2

~~3.3.7 Exceptions to 3.3.6 shall be as follows:~~

<Later> (3.3D)

~~(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.~~ — Later

~~(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure or level) shall be operable.~~ — M15 — L12

<Later> (3.6)

~~(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within days or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours.~~

~~(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 7 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of init loss or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.~~ — Later

< Add 3.5.2, Condition A, Second entry condition > — L9

< Add 3.5.2, Condition B, Second entry condition > — A11

& <LATER> (3.6, 3.7) — <LATER> (3.6) — ~~systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance.~~ — & LATER — L4 — Later

See page 38-1

3.5.2 RA A.1, B.1

3.5.2 RA B.2

& <Later> (3.3D, 3.6, 3.7)

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, ~~3.3.4~~ and ~~3.3.5~~ cannot be met ~~except as noted in 3.3.7 below~~, reactor shutdown shall be initiated ~~and~~ within 72 hours or the reactor shall be in hot shutdown condition MODE 3 within ~~36~~ 6 hours, and, if not corrected, in cold shutdown condition MODE 3 with RCS temperature ≤350°F within an additional ~~72~~ 12 hours. — L7 — M16 — L8 & LATER

3.3.7 Exceptions to 3.3.6 shall be as follows:

<Later> (3.3D)

~~(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.~~ — Later

See page 38-1

(B) If the conditions of Specification 3.3.3(D) cannot be met, reacto operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure level) shall be operable.

<Later> (3.6)

~~(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.~~

~~(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 7 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initiation loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.~~ — Later

< Add SR 3.5.3.1 with Note > M14

< Add 3.5.3 LCO Note > A5

< Add 3.5.3 Condition C > L6

& <LATER> (3.6, 3.7)

systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance. & LATER L4

<LATER> (3.6) Later

See page 38-2

See page 38-1

3.5.3 Condition A

3.5.3 Condition B

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition within 36 hours, and, if not corrected, in cold shutdown condition within an additional 72 hours. 24 within 48 hours MODE 5 A7 A6 & LATER

& <Later> (3.3D, 3.6, 3.7)

3.3.7 Exceptions to 3.3.6 shall be as follows:

<Later> (3.3D)

(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable. Later

See page 38-1

(B) If the conditions of Specification 3.3.3(D) cannot be met, reacto operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure level) shall be operable.

<Later> (3.6)

(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within days or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.

(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 7 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initi loss or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours. Later

<Add 3.5.4 Condition A> M9

<Add 3.5.4 Condition B> M10

See page 38-2, 38-3

systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which woul make the affected system train inoperable for more than 24 consecutiv hours. Prior to initiating maintenance on any component of a train i any system, the redundant component of that system shall be demonstra to be operable within 24 hours prior to the maintenance.

See page 38-3
See page 38-2
See page 38-1

Later (3.6)

3.5.4 RA C.1
3.5.4 RA C.2
& <Later> (3.3D, 3.6, 3.7)

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.3.5 ~~except as noted in 3.3.7 below~~, reactor shutdown shall be cannot be met initiated and the reactor shall be in ~~hot shutdown condition~~ MODE 3 within ~~36~~ 6 hours, and, if not corrected, in ~~cold shutdown condition~~ MODE 5 within an additional ~~72~~ 30 hours.

M9 M10 M11 & LATER

3.3.7 Exceptions to 3.3.6 shall be as follows:

<Later> (3.3D)

~~(A) If the conditions of Specification 3.3.1(F) cannot be met, reacto operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel sh be operable.~~

— Later

(B) If the conditions of Specification 3.3.3(D) cannot be met, reacto operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure level) shall be operable.

See page 38-1

~~(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within days or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.~~

~~(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 7 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of init loss or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.~~

— Later

<Later> (3.6)

& ⟨LATER⟩ (3.5, 3.7) systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance. & LATER L3

⟨LATER⟩ (3.3D, 3.5, 3.7) LATER

3.3.6 If the conditions of Specifications 3.3.1, ~~3.3.2, 3.3.3~~ 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, ~~reactor shutdown shall be initiated and~~ the reactor shall be in ~~hot shutdown~~ condition within ~~36~~ 6 hours, ~~and, if not corrected, in cold shutdown condition within an additional 72 hours.~~ M18 MODE 3 A1 L6 L15

3.6.5 Cond. A
3.6.5 Cond. D
restore in 72 hrs or

3.3.7 ~~Exceptions to 3.3.6 shall be as follows:~~

⟨LATER⟩ (3.3D) ~~(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.~~ LATER

⟨LATER⟩ (3.5) ~~(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure of level) shall be operable.~~ LATER

A6

3.6.5 Cond. B
3.6.5 Cond. D

(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable ~~but both reactor building spray systems are operable~~, restore the inoperable train of cooling to operable status within 7 days or be in at least ~~hot shutdown~~ within the next 6 hours ~~and in cold shutdown within the following 30 hours~~. A1 MODE 3 L5

A6

3.6.5 Cond. C
3.6.5 Cond. D
3.6.5 Cond. B

(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable ~~but both reactor building spray systems are operable~~, restore at least one train of cooling to operable status within 72 hours or be in at least ~~hot shutdown~~ within the next 6 hours ~~and in cold shutdown within the following 30 hours~~. Restore both above required cooling trains to operable status within 7 days of initial loss or be in at least ~~hot shutdown~~ within the next 6 hours ~~and in cold shutdown within the following 30 hours~~. A1 MODE 3 L5

⟨Add 3.6.5 RA A.1 & B.1 CT of 10 days⟩ M18

See Pg. 38-1

systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance.

⟨LATER⟩ (3.3D, 3.5, 3.7) LATER

3.3.6 If the conditions of Specifications 3.3.1, ~~3.3.2, 3.3.3,~~ 3.3.4 and 3.3.5 cannot be met ~~except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition within 36 hours, and,~~ if not corrected, in cold shutdown condition within an additional 72 hours. 36 In 36 hours MODE 5 A7 A1 A10 M19

3.3.7 Exceptions to 3.3.6 shall be as follows:

See Pg. 38-1

(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.

(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure of level) shall be operable.

(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within 7 days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

See Pg. 38-1

systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance. LATER

<LATER> (3.3D, 3.5, 3.7)

3.6.6 Cond.A 3.3.6 If the conditions of Specifications 3.3.1, ~~3.3.2, 3.3.3,~~ 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, ~~reactor shutdown shall be initiated and~~ the reactor shall be in ~~hot shutdown~~ MODE 3 condition within ~~36~~ 6 hours~~, and, if not corrected, in cold shutdown condition within an additional 72 hours.~~

M18 L6 A1 L15

3.6.6 Cond.B restore in 72 hours or

3.3.7 Exceptions to 3.3.6 shall be as follows:

(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.

See Pg. 38-1

(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure of level) shall be operable.

(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within 7 days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

(LATER) (3.6)

(E) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable and one reactor building spray system is inoperable, restore the inoperable spray system to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore the inoperable reactor building emergency cooling train to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

LATER

A2

Bases

The requirements of Specification 3.3.1 assure that below 350°F, adequate long term core cooling is provided. Two low pressure injection pumps are specified. However, only one is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident.

The post-accident reactor building emergency cooling and long-term pressure reduction may be accomplished by two spray units or by a combination of one cooling train and one spray unit. Post-accident iodine removal may be accomplished by one of the two spray system strings. The specified requirements assure that the required post-accident components are available for both reactor building emergency cooling and iodine removal. Specification 3.3.1 assures that the required equipment is operable.

A train consists of two coolers and their associated fans which have sufficient capacity to meet post accident heat removal requirements. Conservatively each reactor building emergency cooling train consists of two fans powered from the same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation.

The borated water storage tank is used for three purposes:

(A) As a supply of borated water for accident conditions.

(B) As an alternate supply of borated water for reaching cold shutdown.(2)

(C) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

(E) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable and one reactor building spray system is inoperable, restore the inoperable spray system to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore the inoperable reactor building emergency cooling train to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

<LATER> (3.6)

LATER

A2

Bases

The requirements of Specification 3.3.1 assure that below 350°F, adequate long term core cooling is provided. Two low pressure injection pumps are specified. However, only one is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident.

The post-accident reactor building emergency cooling and long-term pressure reduction may be accomplished by two spray units or by a combination of one cooling train and one spray unit. Post-accident iodine removal may be accomplished by one of the two spray system strings. The specified requirements assure that the required post-accident components are available for both reactor building emergency cooling and iodine removal. Specification 3.3.1 assures that the required equipment is operable.

A train consists of two coolers and their associated fans which have sufficient capacity to meet post accident heat removal requirements. Conservatively each reactor building emergency cooling train consists of two fans powered from the same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation.

The borated water storage tank is used for three purposes:

(A) As a supply of borated water for accident conditions.

(B) As an alternate supply of borated water for reaching cold shutdown.(4)

(C) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

A14

3.6.5 Cond. B (E) If the conditions of Specification 3.3.4(A) cannot be met because
3.6.5 Cond. A one train of the required reactor building emergency cooling is inoperable and one reactor building spray system is inoperable, restore the inoperable spray system to operable status within 72
3.6.5 Cond. D hours or be in at least ~~hot shutdown~~ MODE 3 A1 within the next 6 hours ~~and in cold shutdown within the following 30 hours~~ L5. Restore the inoperable reactor building emergency cooling train to operable status within 7 days of initial loss or be in at least ~~hot shutdown~~ MODE 3 A1 within the next 6 hours ~~and in cold shutdown within the following 30 hours.~~ L5

~~**Bases**~~ <Add 3.6.5 Cond. G> M20

~~The requirements of Specification 3.3.1 assure that below 350°F, adequate long term core cooling is provided. Two low pressure injection pumps are specified. However, only one is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident.~~

~~The post-accident reactor building emergency cooling and long-term pressure reduction may be accomplished by two spray units or by a combination of one cooling train and one spray unit. Post-accident iodine removal may be accomplished by one of the two spray system strings. The specified requirements assure that the required post-accident components are available for both reactor building emergency cooling and iodine removal. Specification 3.3.1 assures that the required equipment is operable.~~ A2

~~A train consists of two coolers and their associated fans which have sufficient capacity to meet post accident heat removal requirements. Conservatively each reactor building emergency cooling train consists of two fans powered from the same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation.~~

~~The borated water storage tank is used for three purposes:~~

~~(A) As a supply of borated water for accident conditions.~~

~~(B) As an alternate supply of borated water for reaching cold shutdown.(2)~~

~~(C) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)~~

<LATER> (3.6)

(E) If the conditions of Specification 3.3.4(A) cannot be met becaus one train of the required reactor building emergency cooling is inoperable and one reactor building spray system is inoperable, restore the inoperable spray system to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore the inoper reactor building emergency cooling train to operable status with days of initial loss or be in at least hot shutdown within the n 6 hours and in cold shutdown within the following 30 hours.

LATER

A2

Bases

The requirements of Specification 3.3.1 assure that below 350°F, adequate lo term core cooling is provided. Two low pressure injection pumps are specifi However, only one is necessary to supply emergency coolant to the reactor in event of a loss-of-coolant accident.

The post-accident reactor building emergency cooling and long-term pressure reduction may be accomplished by two spray units or by a combination of one cooling train and one spray unit. Post-accident iodine removal may be accomplished by one of the two spray system strings. The specified requirem assure that the required post-accident components are available for both rea building emergency cooling and iodine removal. Specification 3.3.1 assures the required equipment is operable.

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The borated water storage tank is used for three purposes:

(A) As a supply of borated water for accident conditions.

(B) As an alternate supply of borated water for reaching cold shutdown.(2)

(C) As a supply of borated water for flooding the fuel transfer cana during refueling operation.(3)

370,100 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. Approximately 16,000 gallons of borated water are required to reach cold shutdown. The original nominal borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The minimum required BWST boron concentration of 2270 ppm assures that the core will be maintained at least 1 percent $\Delta k/k$ subcritical at 70°F without any control rods in the core.

A2

Specification 3.3.2 assures that above 350°F two high pressure injection pumps are also available to provide injection water as the energy of the reactor coolant system is increased.

Specification 3.3.3 assures that above 800 psig both core flooding tanks are operational. Since their design pressure is 600 ± 25 psig, they are not brought into the operational state until 800 psig to prevent spurious injection of borated water. Both core flooding tanks are specified as a single core flood tank has insufficient inventory to reflood the core.(1)

Specification 3.3.4 assures that prior to going critical the redundant train of reactor building emergency cooling and spray train are operable.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

The volume specified by 3.3.4.B is the safety analysis volume and does not contain allowances for instrument uncertainty. 9,000 gallons corresponds to a level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. No maximum volume is specified as the value used as the maximum volume in the safety analysis bounds the physical size of the NaOH tank. Additional allowances for instrument uncertainties, as determined in Reference 6, are incorporated in the operating procedures associated with the level instrumentation used in the control room.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2200°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.(4)

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

A15

370,100 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. Approximately 16,000 gallons of borated water are required to reach cold shutdown. The original nominal borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The minimum required BWST boron concentration of 2270 ppm assures that the core will be maintained at least 1 percent Δk/k subcritical at 70°F without any control rods in the core.

A2

Specification 3.3.2 assures that above 350°F two high pressure injection pumps are also available to provide injection water as the energy of the reactor coolant system is increased.

Specification 3.3.3 assures that above 800 psig both core flooding tanks are operational. Since their design pressure is 600 ± 25 psig, they are not brought into the operational state until 800 psig to prevent spurious injection of borated water. Both core flooding tanks are specified as a single core flood tank has insufficient inventory to reflood the core.(1)

Specification 3.3.4 assures that prior to going critical the redundant train of reactor building emergency cooling and spray train are operable.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

The volume specified by 3.3.4.B is the safety analysis volume and does not contain allowances for instrument uncertainty. 9,000 gallons corresponds to a level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. No maximum volume is specified as the value used as the maximum volume in the safety analysis bounds the physical size of the NaOH tank. Additional allowances for instrument uncertainties, as determined in Reference 6, are incorporated in the operating procedures associated with the level instrumentation used in the control room.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2200°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.(4)

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

3.6.5

370,100 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. Approximately 16,000 gallons of borated water are required to reach cold shutdown. The original nominal borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The minimum required BWST boron concentration of 2270 ppm assures that the core will be maintained at least 1 percent Δk/k subcritical at 70°F without any control rods in the core. A2

Specification 3.3.2 assures that above 350°F two high pressure injection pumps are also available to provide injection water as the energy of the reactor coolant system is increased.

Specification 3.3.3 assures that above 800 psig both core flooding tanks are operational. Since their design pressure is 600 ± 25 psig, they are not brought into the operational state until 800 psig to prevent spurious injection of borated water. Both core flooding tanks are specified as a single core flood tank has insufficient inventory to reflood the core.(1)

Specification 3.3.4 assures that prior to going critical the redundant train of reactor building emergency cooling and spray train are operable.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

The volume specified by 3.3.4.B is the safety analysis volume and does not contain allowances for instrument uncertainty. 9,000 gallons corresponds to a level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. No maximum volume is specified as the value used as the maximum volume in the safety analysis bounds the physical size of the NaOH tank. Additional allowances for instrument uncertainties, as determined in Reference 6, are incorporated in the operating procedures associated with the level instrumentation used in the control room.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2200°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.(4)

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

3.7.7

370,100 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. Approximately 16,000 gallons of borated water are required to reach cold shutdown. The original nominal borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The minimum required BWST boron concentration of 2270 ppm assures that the core will be maintained at least 1 percent Δk/k subcritical at 70°F without any control rods in the core.

A2

Specification 3.3.2 assures that above 350°F two high pressure injection pumps are also available to provide injection water as the energy of the reactor coolant system is increased.

Specification 3.3.3 assures that above 800 psig both core flooding tanks are operational. Since their design pressure is 600 ± 25 psig, they are not brought into the operational state until 800 psig to prevent spurious injection of borated water. Both core flooding tanks are specified as a single core flood tank has insufficient inventory to reflood the core.(1)

Specification 3.3.4 assures that prior to going critical the redundant train of reactor building emergency cooling and spray train are operable.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

The volume specified by 3.3.4.B is the safety analysis volume and does not contain allowances for instrument uncertainty. 9,000 gallons corresponds to a level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. No maximum volume is specified as the value used as the maximum volume in the safety analysis bounds the physical size of the NaOH tank. Additional allowances for instrument uncertainties, as determined in Reference 6, are incorporated in the operating procedures associated with the level instrumentation used in the control room.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2200°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.(4)

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

A12

3.3.15

REFERENCES

(1) FSAR, Section 14.2.5

(2) FSAR, Section 3.2

(3) FSAR, Section 9.5.2

(4) FSAR, Section 9.3.1

(5) FSAR, Section 6.3

(6) ANO Calculation 91-E-0019-01

A2

~~**REFERENCES**~~

~~(1) FSAR, Section 14.2.5~~

~~(2) FSAR, Section 3.2~~

~~(3) FSAR, Section 9.5.2~~

~~(4) FSAR, Section 9.3.1~~

~~(5) FSAR, Section 6.3~~

~~(6) ANO Calculation 91-E-0019-01~~

A2

REFERENCES

(1) FSAR, Section 14.2.5

(2) FSAR, Section 3.2

(3) FSAR, Section 9.5.2

(4) FSAR, Section 9.3.1

(5) FSAR, Section 6.3

(6) ANO Calculation 91-E-0019-01

A2

3.4.4
3.4.5

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the turbine cycle components for removal of reactor decay heat.

Objective

To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.

Specifications

MODES 1,2,3 — A1 & LATER

3.4.4 APPL
3.4.5 APPL
& <LATER> (3.7)

3.4.1 ~~The reactor shall not be heated above 280°F~~ unless the following conditions are met:

3.4.4 LCO
3.4.5 LCO

1. Capability to remove decay heat by use of two steam generators. — LA1 BASES

<LATER> (3.7) — LATER

2. ~~Fourteen of the steam system safety valves are operable.~~

3. ~~A minimum usable volume of 32,300 gallons of water is available in Tank T41B.~~

4. ~~(Deleted)~~

5. ~~Both main steam block valves and both main feedwater isolation valves are operable.~~

3.4.4 RA B.1
3.4.5 RA A.1, B.1
& <LATER> (3.7)

3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the hot shutdown condition within 12 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours. — M9, L6 & LATER

<LATER> (3.7) — LATER

3.4.3 Two (2) EFW trains shall be operable as follows:

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<LATER> (3.7) — LATER

* Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

A18

3.4 STEAM AND POWER CONVERSION SYSTEM

~~Applicability~~ (A1)

~~Applies to the turbine cycle components for removal of reactor decay heat.~~

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

3.7.1 APPL & <LATER> (3.4A) — 3.4.1 ~~The reactor shall not be heated above 280°F unless the following conditions are met:~~ (A1) & LATER

~~1. Capability to remove decay heat by use of two steam generators.~~ LATER (L1)

3.7.1 LCO — *2. Fourteen ~~of the steam system~~ safety valves are operable. (M2)

see pg 40-5 — 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B. (A1)

~~4. (Deleted)~~ <ADD TABLE 3.7.1-1> (L1)

see pg 40-2 & see pg 40-3 — 5. Both main steam block valves and both main feedwater isolation valves are operable.

{Reduce power per Table 3.7.1-1 -- 4 hours} (M24)

3.7.1 RA A.1, 3.7.1 RA B.1, B.2 & <LATER> (3.4A) — 3.4.2 ~~Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours.~~ (L1) ~~If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours,~~ the reactor shall be placed in the hot shutdown MODE 3 (A1) condition within 12 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition MODE 4 within 24 hours. (M1)

3.4.3 Two (2) EFW trains ~~shall be~~ operable as follows: (L3)

See pg 40-4 —

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<ADD 3.7.1 ACTIONS Note> (L1)

<Add 3.7.1 RA A.2> (M24)

3.7.1 LCO NOTE, SR 3.7.1.1 NOTE 2 — * Except ~~that~~ during hydrotests, with the reactor subcritical, ~~fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.~~ (LA1) BASES

See pg 40-4 — ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

3.7.2

3.4 STEAM AND POWER CONVERSION SYSTEM

~~Applicability~~

~~Applies to the turbine cycle components for removal of reactor decay heat.~~ A1

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

{MODE 1,2 & 3}

3.7.2 APPL & <LATER> (3.4A) 3.4.1 The reactor shall not be heated ~~above 280°F unless the following~~ conditions are met: A1 & LATER

~~1. Capability to remove decay heat by use of two steam generators.~~ LATER

see pg 40-1 *2. Fourteen of the steam system safety valves are operable.

see pg 40-5 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

~~4. (Deleted)~~ A1

3.7.2 LCO 5. Both main steam block valves ~~and both~~ main feedwater isolation valves are operable.

see pg 40-3

3.7.2 RA A.1, B.1, C.1, D.1 3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the ~~hot shutdown~~ MODE 3 condition within 12 hours. ~~If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.~~ A1 L4

& <LATER> (3.4A)

MODE 4 C3

see pg 40-4 3.4.3 Two (2) EFW trains shall be operable as follows:

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<Add 3.7.2 Cond C Note> L4

<Add 3.7.2 RA C.2> M4

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see pg 40-1 * Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

see pg 40-4 ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

A14

3.4 STEAM AND POWER CONVERSION SYSTEM

~~Applicability~~ A1

~~Applies to the turbine cycle components for removal of reactor decay heat.~~

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

3.7.3 APPL & <LATER> (3.4A)

3.4.1 ~~The reactor shall not be heated above 280°F unless the following conditions are met:~~ MODES 1, 2, & 3 A1 & LATER

1. Capability to remove decay heat by use of two steam generators. LATER

See pg 40-1 — 2. Fourteen of the steam system safety valves are operable.

See pg 40-5 — 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

~~4. (Deleted)~~ A1

See pg 40-2

3.7.3 LCO — 5. Both main steam block valves and both main feedwater isolation valves are operable.

3.7.3 A.1, B.1, C.1, D.1 & <LATER> (3.4A)

3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the hot shutdown MODE 3 A1 condition within 12 hours. ~~If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.~~ L4 MODE 4 L3

3.4.3 Two (2) EFW trains shall be operable as follows:

See pg 40-4

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<Add 3.7.3 Cond C Note> L4

<Add 3.7.3 RA C.2> M4

See pg 40-1 — * Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

See pg 40-4 — ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

A14

3.7.5

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the turbine cycle components for removal of reactor decay heat.

A1

Objective

To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.

Specifications

See pgs 40-1,2,3,5 3.4.1 The reactor shall not be heated above 280°F unless the following conditions are met:

<LATER> (3.4A) 1. ~~Capability to remove decay heat by use of two steam generators.~~ LATER

See pg 40-1 *2. Fourteen of the steam system safety valves are operable.

See pg 40-5 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

~~4. (Deleted)~~ A1

See pgs 40-2,3 5. Both main steam block valves and both main feedwater isolation valves are operable.

3.4.2 See pgs 40-1,2,3,5 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the hot shutdown condition within 12 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.

3.7.5 LCO 3.4.3 Two (2) EFW trains shall be operable as follows:

3.7.5 APPL Note 1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

3.7.5 APPL 2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

See pg 40-1 * Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

SR 3.7.5.2 Note ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

A14

3.7.6

3.4 STEAM AND POWER CONVERSION SYSTEM

~~Applicability~~

~~Applies to the turbine cycle components for removal of reactor decay heat.~~ (A1)

~~Objective~~

~~To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.~~

Specifications

<MODES 1, 2, 3, & 4 when rely on SG>

3.7.6 APPL & <LATER> (3.4A) 3.4.1 The reactor shall ~~not~~ be heated above 280°F ~~unless the following conditions are met:~~ (M6) & LATER

1. ~~Capability to remove decay heat by use of two steam generators.~~ LATER

See pg 40-1 2. Fourteen of the steam system safety valves are operable.*

3.7.6 LCO 3. A minimum usable volume of 32,300 gallons of water is available in Tank T41B.

~~4. (Deleted)~~ (A1)

See pg 40-2,3 5. Both main steam block valves and both main feedwater isolation valves are operable.

3.7.6 A.2, 3.7.6 B.1, B.2 & <LATER> (3.4A) 3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the hot shutdown condition within ~~12~~ 6 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours. (L7) & LATER (M7) (L7)

See pg 40-4 3.4.3 Two (2) EFW trains shall be operable as follows:

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.

2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<Add 3.7.6 RA A.1> (M7)

<Add SR 3.7.6.1> (M7)

See pg 40-1 * Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

See pg 40-4 ** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

(A14)

3.4.4 If the conditions specified in 3.4.3 cannot be met:

3.7.5 RA E.1

1. With the motor driven EFW pump or its associated flow path inoperable and RCS conditions above CSD and RCS temperature < 280°F and any Steam Generator relied upon for heat removal, immediately initiate action to restore the EFW train to operable status.

3.7.5 RA A.1
3.7.5 RA C.1
3.7.5 RA C.2

2. With the RCS temperature ≥ 280°F and one steam generator supply path to the turbine driven EFW pump inoperable, restore the steam generator supply path to operable status within 7 days or be in Hot Shutdown within 6 hours and reduce RCS temperature to < 280°F within the next 12 hours.

3.7.5 RA B.1
3.7.5 RA C.1
3.7.5 RA C.2

3. With the RCS temperature ≥ 280°F and one EFW pump or its associated flow path inoperable, restore the EFW train to operable status within 72 hours or be in Hot Shutdown within 6 hours, and reduce RCS temperature to < 280°F within the next 12 hours.

4. ~~With the RCS temperature ≥ 280°F, both EFW pumps or their associated flow paths inoperable, and the Auxiliary Feedwater pump available, be in Hot Shutdown within 6 hours, and reduce RCS temperature to < 280°F within the next 12 hours.~~ L8

3.7.5 RA D.1
3.7.5 RA D.1 Note

5. With the RCS temperature ≥ 280°F and both EFW pumps or their associated flow paths inoperable, ~~and the Auxiliary Feedwater pump unavailable,~~ immediately initiate action to restore one EFW train ~~the Auxiliary Feedwater pump~~ to operable status. LCO 3.0.3 and all other LCO Required Actions requiring mode changes are suspended until one EFW train ~~or the Auxiliary Feedwater pump~~ is restored to operable status.

< Add 10 day Completion Time for 3.7.5 RA A.1 and RA B.1 > M8

3.7.5

A2

Bases

The Emergency Feedwater (EFW) system is designed to provide flow sufficient to remove heat load equal to 3½ percent full power operation. The system minimum flow requirement to the steam generator(s) is 500 gpm. This takes into account a single failure, pump recirculation flow, seal leakage and pump wear.

In the event of loss of main feedwater, feedwater is supplied by the emergency feedwater pumps, one which is powered from an operable emergency bus and one which is powered from an operable steam supply system. Both EFW pumps take suction from tank T41B. Decay heat is removed from a steam generator by steam relief through the turbine bypass, atmospheric dump valves, or safety valves. Fourteen of the steam safety valves will relieve the necessary amount of steam for rated reactor power.

The EFW system is considered to be operable when the components and flow paths required to provide EFW flow to the steam generators are operable. This requires that the turbine driven EFW pump be operable with redundant steam supplies from each of the main steam lines upstream of the MSIVs (CV-2617 and CV-2667) and capable of supplying EFW flow to either of the tv steam generators. The motor driven EFW pump and associated flow path to th EFW system is also required to be operable. The piping, valves, instrumentation, and controls in the required flow paths shall also be operable. One EFW train, which includes the motor driven EFW pump, is required to be operable when above CSD and below 280°F with any steam generator relied upon for heat removal. This is because of reduced heat removal requirements, the short duration EFW would be required, and the insufficient steam supply available in this condition to power the turbine driven EFW pump.

When one of the required EFW trains is inoperable, action must be taken to restore the train to operable status within 72 hours. This condition includes loss of the steam supply to the turbine driven EFW pump. The 72 hour completion time is reasonable, based on the redundant capabilities afforded by the EFW system, time needed for repairs, and the low probabili of a DBA occurring during this time period.

With two EFW trains inoperable, the unit must be placed in a mode in which the LCO does not apply using the Auxiliary Feedwater pump. With RCS temperature < 280°F the Decay Heat Removal system may be placed in operatic

With both EFW trains inoperable and the Auxiliary Feedwater pump unavailab the unit is in a seriously degraded condition with only limited means for conducting a cooldown using nonsafety grade equipment. In such a conditio the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires t action be started immediately to restore at least one EFW pump or the Auxi Feedwater pump to Operable status. LCO 3.0.3 is not applicable, as it cou force the unit into a less safe condition.

3.7.6

~~The OPERABILITY of the condensate storage tank with the minimum required water volume ensures that sufficient water is available to support EFW operation for both units for at least 30 minutes. This provides adequate time for the operators to manually switch the EFW suction alignment to the Service Water System (SWS), if required. The SWS provides the assured long-term source of cooling water. The required volume considers that the EFWS of both units may be aligned to T41B simultaneously. The tank is seismically qualified and the required volume is also protected from tornado missiles.~~

~~The required minimum usable volume includes an allowance for losses due to Unit 2 recirculation line flow. It does not include any allowance for instrument uncertainty or for the unusable volume due to the suction piping configuration. This volume is equivalent to a tank level of 3'-10".~~

~~The tank has sufficient capacity to support more than four hours of cooling flow for both units. This capability is not considered to be a safety related design function and is not controlled by the Technical Specifications.~~

A2

414

LAR

3.3.1
3.3.4
3.3.10

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objectives

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

A1

3.5.1.1 Startup and operation are not permitted unless the requirements Table 3.5.1-1, columns 3 and 4 are met.

& (LATER) (3.3B, 3.3C, 3.3D, 3.4B)

3.5.1.2 In the event the number of protection channels operable falls be the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

A3 & LATER

3.5.1.3 For on-line testing or in the event of a protection instrument or channel failure, a key operated channel bypass switch associated with each reactor protection channel may be used to lock the channel trip relay in the untripped state as indicated by a light.

LA1 BASES

3.3.1 RA B.1

Only one channel shall be locked in the untripped state or contain inoperable functions in the untripped state at any one time. In the event more than one protection channel contains inoperable functions in the untripped state, or a protection channel or function becomes inoperable concurrent with another protection channel locked in the untripped state, within 1 hour implement the actions required by Table 3.5.1-1 Note 6.

A3

Only one channel bypass key shall be accessible for use in the control room.

LA1 BASES

3.3.1 RA A.1 & A.2
3.3.1 RA B.2.1 & B.2.2

While operating with an inoperable function unbypassed in the untripped state, the remaining RPS key operated channel bypass switches shall be tagged to prevent their operation.

M13

3.5.1.4 The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation except during channel testing.

R TRM

SR 3.3.10.4

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade.

A1

3.3.10 RA A.1 & A.2

If the overlap is less than one decade, the flux level shall be maintained in the source range until the one decade overlap is achieved.

immediately suspend positive reactivity additions and open CRD trip breakers within 1 hr.

M10

3.3.4 RA B.2

3.5.1.6 In the event that one of the trip devices in either of the source supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within 30 minutes following detection.

1 hr

The condition will be corrected and the remaining trip devices shall be tested within eight hours following detection. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour period the reactor shall be placed in the hot shutdown condition within an additional four hours.

L1

3.3.5
3.3.6
3.3.7

A1

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objectives

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

A3

<LATER> (3.3A, 3.3C, 3.3D, 3.4B)

3.5.1.1 Startup and operation are not permitted unless the requirements Table 3.5.1-1, columns 3 and 4 are met.

3.5.1.2 In the event the number of protection channels operable falls be the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

& LATER

<LATER> (3.3A)

3.5.1.3 For on-line testing or in the event of a protection instrument c channel failure, a key operated channel bypass switch associatec with each reactor protection channel may be used to lock the channel trip relay in the untripped state as indicated by a ligh Only one channel shall be locked in the untripped state or contain inoperable functions in the untripped state at any one time. In the event more than one protection channel contains inoperable functions in the untripped state, or a protection channel or function becomes inoperable concurrent with another protection channel locked in the untripped state, within 1 hour implement the actions required by Table 3.5.1-1 Note 6. Only or channel bypass key shall be accessible for use in the control r: While operating with an inoperable function unbypassed in the untripped state, the remaining RPS key operated channel bypass switches shall be tagged to prevent their operation.

LATER

3.5.1.4 The key operated shutdown bypass switch associated with each rea protection channel shall not be used during reactor power operat except during channel testing.

R
TRM

<LATER> (3.3A)

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall be mainta: in the source range until the one decade overlap is achieved.

3.5.1.6 In the event that one of the trip devices in either of the sour: supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within minutes following detection. The condition will be corrected ar the remaining trip devices shall be tested within eight hours following detection. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour per: the reactor shall be placed in the hot shutdown condition withi: additional four hours.

LATER

3.3.11
3.3.12
3.3.13

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

A1

Objectives

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

A3

& LATER (3.3A, 3.3B, 3.3D, 3.4B)

& LATER

3.5.1.1 Startup and operation are not permitted unless the requirements Table 3.5.1-1, columns 3 and 4 are met.

3.5.1.2 In the event the number of protection channels operable falls be the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

LATER (3.3A)

– LATER

3.5.1.3 For on-line testing or in the event of a protection instrument channel failure, a key operated channel bypass switch associated with each reactor protection channel may be used to lock the channel trip relay in the untripped state as indicated by a ligh Only one channel shall be locked in the untripped state or contain inoperable functions in the untripped state at any one time. In the event more than one protection channel contains inoperable functions in the untripped state, or a protection channel or function becomes inoperable concurrent with another protection channel locked in the untripped state, within 1 hour implement the actions required by Table 3.5.1-1 Note 6. Only one channel bypass key shall be accessible for use in the control ro While operating with an inoperable function unbypassed in the untripped state, the remaining RPS key operated channel bypass switches shall be tagged to prevent their operation.

3.5.1.4 The key operated shutdown bypass switch associated with each rea protection channel shall not be used during reactor power operat except during channel testing.

R TRM

– LATER

LATER (3.3A)

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall be maintai in the source range until the one decade overlap is achieved.

3.5.1.6 In the event that one of the trip devices in either of the sour supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within minutes following detection. The condition will be corrected a: the remaining trip devices shall be tested within eight hours following detection. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour per: the reactor shall be placed in the hot shutdown condition withi: additional four hours.

3.3.15
3.3.16

3.5 INSTRUMENTATION SYSTEMS

3.5.1 <u>Operational Safety Instrumentation</u>

<u>Applicability</u>

Applies to unit instrumentation and control systems.

<u>Objectives</u>

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

<u>Specifications</u>

A1

3.5.1.1 Startup and operation are not permitted unless the requirements Table 3.5.1-1, columns 3 and 4 are met.

3.5.1.2 In the event the number of protection channels operable falls be the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

A3
& Later
& <Later> (3.3A, 3.3B, 3.3C, 3.4B)

3.5.1.3 For on-line testing or in the event of a protection instrument c channel failure, a key operated channel bypass switch associatec with each reactor protection channel may be used to lock the channel trip relay in the untripped state as indicated by a ligh Only one channel shall be locked in the untripped state or contain inoperable functions in the untripped state at any one time. In the event more than one protection channel contains inoperable functions in the untripped state, or a protection channel or function becomes inoperable concurrent with another protection channel locked in the untripped state, within 1 hour implement the actions required by Table 3.5.1-1 Note 6. Only or channel bypass key shall be accessible for use in the control rc While operating with an inoperable function unbypassed in the untripped state, the remaining RPS key operated channel bypass switches shall be tagged to prevent their operation.

<LATER> (3.3A)
LATER

3.5.1.4 The key operated shutdown bypass switch associated with each rea protection channel shall not be used during reactor power opera except during channel testing.

R
TRM

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall be mainta: in the source range until the one decade overlap is achieved.

3.5.1.6 In the event that one of the trip devices in either of the sour: supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within minutes following detection. The condition will be corrected ar the remaining trip devices shall be tested within eight hours following detection. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour per: the reactor shall be placed in the hot shutdown condition withi: additional four hours.

<LATER> (3.3A)
LATER

3.4.14

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objectives

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

A1

3.5.1.1 Startup and operation are not permitted unless the requirements of Table 3.5.1-1, columns 3 and 4 are met.

<LATER> (3.3A, 3.3B, 3.3C, 3.3D)

3.5.1.2 In the event the number of protection channels operable falls below the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

A10 & LATER

<LATER> (3.3A)

3.5.1.3 For on-line testing or in the event of a protection instrument or channel failure, a key operated channel bypass switch associated with each reactor protection channel may be used to lock the channel trip relay in the untripped state as indicated by a light. Only one channel shall be locked in the untripped state or contain inoperable functions in the untripped state at any one time. In the event more than one protection channel contains inoperable functions in the untripped state, or a protection channel or function becomes inoperable concurrent with another protection channel locked in the untripped state, within 1 hour implement the actions required by Table 3.5.1-1 Note 6. Only one channel bypass key shall be accessible for use in the control room. While operating with an inoperable function unbypassed in the untripped state, the remaining RPS key operated channel bypass switches shall be tagged to prevent their operation.

LATER

3.5.1.4 The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation except during channel testing.

R TRM

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall be maintained in the source range until the one decade overlap is achieved.

<LATER> (3.3A)

3.5.1.6 In the event that one of the trip devices in either of the sources supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within 30 minutes following detection. The condition will be corrected and the remaining trip devices shall be tested within eight hours following detection. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour period, the reactor shall be placed in the hot shutdown condition within an additional four hours.

LATER

Items on this page also addressed in the following packages: 3.3 A, 3.3 B, 3.3 C, 3.3 D, 3.4 B

SR

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objectives

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

3.5.1.1 Startup and operation are not permitted unless the requirements of Table 3.5.1-1, columns 3 and 4 are met.

3.5.1.2 In the event the number of protection channels operable falls below the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

3.5.1.3 For on-line testing or in the event of a protection instrument or channel failure, a key operated channel bypass switch associated with each reactor protection channel may be used to lock the channel trip relay in the untripped state as indicated by a light. Only one channel shall be locked in the untripped state or contain inoperable functions in the untripped state at any one time. In the event more than one protection channel contains inoperable functions in the untripped state, or a protection channel or function becomes inoperable concurrent with another protection channel locked in the untripped state, within 1 hour implement the actions required by Table 3.5.1-1 Note 6. Only one channel bypass key shall be accessible for use in the control room. While operating with an inoperable function unbypassed in the untripped state, the remaining RPS key operated channel bypass switches shall be tagged to prevent their operation.

3.5.1.4 ~~The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation except during channel testing.~~ R TRM

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall be maintained in the source range until the one decade overlap is achieved.

3.5.1.6 In the event that one of the trip devices in either of the sources supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within 30 minutes following detection. The condition will be corrected and the remaining trip devices shall be tested within eight hours following detection. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour period, the reactor shall be placed in the hot shutdown condition within an additional four hours.

3.5.1.7 The Decay Heat Removal System isolation valve closure setpoints shall be equal to or less than 340 psig for one valve and equal to or less than 400 psig for the second valve in the suction line. The relief valve setting for the DHR system shall be equal to or less than 450 psig. — LATER

⟨LATER⟩ (3.4B)

3.5.1.8 The degraded voltage monitoring relay settings shall be as follows: — LATER

⟨LATER⟩ (3.3D)

a. The 4.16 KV emergency bus undervoltage relay setpoints shall be >3115 VAC but <3177 VAC.

b. The 460 V emergency bus undervoltage relay setpoints shall be >423 VAC but <431 VAC with a time delay setpoint of 8 seconds ± 1 second.

3.5.1.9 The following Reactor Trip circuitry shall be operable as indicated: — A1

Table 3.3.1-1, Function 10, "Applicable MODES"

1. Reactor trip upon loss of Main Feedwater shall be operable (as determined by Specification 4.1.a and item 35 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 10% reactor power.) — A1, L3

Table 3.3.1-1, Function 9 "Applicable MODES"

2. Reactor trip upon Turbine Trip shall be operable (as determined by Specification 4.1.a and item 41 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 45% reactor power.) — A1, L3

Table 3.3.1-1, Cond. ref. from RA C.1 Function 10 - 3.3.1 RA G.1 Function 9 - 3.3.1 RA F.1

3. If the requirements of Specifications 3.5.1.9.1 or 3.5.1.9.2 cannot be met, restore the inoperable trip within 12 hours or bring the plant to a hot shutdown condition. — cannot be met, reduce power to ≤ 10% RTP, — reduce power to ≤ 45% RTP — L3

3.5.1.10 Deleted — A1

3.5.1.11 For on-line testing of the Emergency Feedwater Initiation and Control (EFIC) system channels during power operation only one channel shall be locked into "maintenance bypass" at any one time. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. — LATER

⟨LATER⟩ (3.3C)

3.5.1.12 The Containment High Range Radiation Monitoring Instrumentation shall be operable with a minimum measurement range from 1 to 10^7 R/hr. — LATER

⟨LATER⟩ (3.3D)

<LATER> (3.4B)

3.5.1.7 The Decay Heat Removal System isolation valve closure setpoints shall be equal to or less than 340 psig for one valve and equal to or less than 400 psig for the second valve in the suction line. The relief valve setting for the DHR system shall be equal to or less than 450 psig. — LATER

<LATER> (3.3D)

3.5.1.8 The degraded voltage monitoring relay settings shall be as follows: — LATER

a. The 4.16 KV emergency bus undervoltage relay setpoints shall be >3115 VAC but <3177 VAC.

b. The 460 V emergency bus undervoltage relay setpoints shall be >423 VAC but <431 VAC with a time delay setpoint of 8 seconds ±1 second.

<LATER> (3.3A)

3.5.1.9 The following Reactor Trip circuitry shall be operable as indicated: — LATER

1. Reactor trip upon loss of Main Feedwater shall be operable (as determined by Specification 4.1.a and item 35 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 10% reactor power.)

2. Reactor trip upon Turbine Trip shall be operable (as determined by Specification 4.1.a and item 41 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 45% reactor power.)

3. If the requirements of Specifications 3.5.1.9.1 or 3.5.1.9.2 cannot be met, restore the inoperable trip within 12 hours or bring the plant to a hot shutdown condition.

~~3.5.1.10 Deleted~~ — A1 |

3.5.1.11 For on-line testing of the Emergency Feedwater Initiation and Control (EFIC) system channels during power operation only one channel shall be locked into "maintenance bypass" at any one time. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. — LA1 Bases

<LATER> (3.3D)

3.5.1.12 The Containment High Range Radiation Monitoring Instrumentation shall be operable with a minimum measurement range from 1 to 10^7 R/hr. — LATER

3.3.8
3.3.15

3.5.1.7 〈LATER〉 (3.4B) ~~The Decay Heat Removal System isolation valve closure setpoints shall be equal to or less than 340 psig for one valve and equal to or less than 400 psig for the second valve in the suction line. The relief valve setting for the DHR system shall be equal to or less than 450 psig.~~ — LATER

3.5.1.8 SR 3.3.8.2 The degraded voltage monitoring relay settings shall be as follows:

a. The 4.16 KV emergency bus undervoltage relay setpoints shall be >3115 VAC but <3177 VAC.

b. The 460 V emergency bus undervoltage relay setpoints shall be >423 VAC but <431 VAC with a time delay setpoint of 8 seconds ±1 second.

3.5.1.9 〈LATER〉 (3.3A) ~~The following Reactor Trip circuitry shall be operable as indicated:~~ — LATER

~~1. Reactor trip upon loss of Main Feedwater shall be operable (as determined by Specification 4.1.a and item 35 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 10% reactor power.)~~

~~2. Reactor trip upon Turbine Trip shall be operable (as determined by Specification 4.1.a and item 41 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 45% reactor power.)~~

~~3. If the requirements of Specifications 3.5.1.9.1 or 3.5.1.9.2 cannot be met, restore the inoperable trip within 12 hours or bring the plant to a hot shutdown condition.~~

3.5.1.10 ~~Deleted~~ A1

3.5.1.11 〈LATER〉 (3.3C) ~~For on-line testing of the Emergency Feedwater Initiation and Control (EFIC) system channels during power operation only one channel shall be locked into "maintenance bypass" at any time. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed.~~ — LATER

3.5.1.12 Table 3.3.15-1 PAM #9 The Containment High Range Radiation Monitoring instrumentation shall be operable ~~with a minimum measurement range from 1 to 10^7 R/hr.~~ LA1 Bases

3.5.1.7 SR 3.4.14.3 The Decay Heat Removal System isolation valve closure setpoints shall be equal to or less than 340 psig for one valve and equal to or less than 400 psig for the second valve in the suction line. ~~The relief valve setting for the DHR system shall be equal to or less than 450 psig.~~ LA1 BASES

3.5.1.8 ~~The degraded voltage monitoring relay settings shall be as follows:~~ <LATER> (3.3D) LATER

~~a. The 4.16 KV emergency bus undervoltage relay setpoints shall be >3115 VAC but <3177 VAC.~~

~~b. The 460 V emergency bus undervoltage relay setpoints shall be >423 VAC but <431 VAC with a time delay setpoint of 8 seconds ±1 second.~~

3.5.1.9 ~~The following Reactor Trip circuitry shall be operable as indicated:~~ <LATER> (3.3 A) LATER

~~1. Reactor trip upon loss of Main Feedwater shall be operable (as determined by Specification 4.1.a and item 35 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 10% reactor power.)~~

~~2. Reactor trip upon Turbine Trip shall be operable (as determined by Specification 4.1.a and item 41 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 45% reactor power.)~~

~~3. If the requirements of Specifications 3.5.1.9.1 or 3.5.1.9.2 cannot be met, restore the inoperable trip within 12 hours or bring the plant to a hot shutdown condition.~~

3.5.1.10 Deleted A1

3.5.1.11 ~~For on-line testing of the Emergency Feedwater Initiation and Control (EFIC) system channels during power operation only one channel shall be locked into "maintenance bypass" at any one time. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed.~~ <LATER> (3.3C) LATER

3.5.1.12 ~~The Containment High Range Radiation Monitoring instrumentation shall be operable with a minimum measurement range from 1 to 10^7 R/hr.~~ <LATER> (3.3 D) LATER

<LATER> (3.3D)

3.5.1.13 ~~Two control room ventilation radiation monitoring channels shall be operable whenever the reactor coolant system is above the cold shutdown condition or during handling of irradiated fuel.~~ LATER

<LATER> (3.3D)

3.5.1.14 ~~The Main Steam Line Radiation Monitoring Instrumentation shall be operable with a minimum measurement range from 10^{-1} to 10^{4} mR/hr, whenever the reactor is above the cold shutdown condition.~~ LATER

Table 3.3.11.1

3.5.1.15 Initiate functions of the EFIC system ~~which are bypassed at cold shutdown conditions~~ shall have the following minimum operability conditions: A13

#1.c Appl w/ Note (a) — a. "low steam generator pressure" initiate shall be operable when the main steam pressure ~~exceeds~~ ≥ 750 psig. A8

#1.d Appl — b. "loss of 4 RC pumps" initiate shall be operable when neutron flux ~~exceeds~~ ≥ 10% power. A9

#1.a Appl — c. "main feedwater pumps tripped" initiate shall be operable when neutron flux ~~exceeds~~ ≥ 10% power.

#3.a Appl w/ Note (a) + <LATER> (3.7)

3.5.1.16 The automatic steam generator isolation system within EFIC shall be operable when main steam pressure ~~is greater than~~ ≥ 750 psig. +LATER A8

<Add 3.3.11 RA F.2.1 for Table 3.3.11-1, Function 1.c> M9

<Add Appl. for Table 3.3.11-1, Function 1.b>
<Add 3.3.11 RA D.2 for Table 3.3.11-1, Function 1.b> M2

<Add LCO and Appl for Table 3.3.11-1, Functions 2.a and 2.b, + Table Note (a)>
<Add 3.3.11 RA C.1, F.1 + F.2.1 for Table 3.3.11-1, Functions 2.a and 2.b> M3

<Add Table 3.3.11-1, Function 3.a, Note (b)>
<Add 3.3.11 RA F.2.1 + F.2.2 w/Note for Table 3.3.11-1, Function 3.a> M4

3.3.16 LCO
3.3.16 Appl.

3.5.1.13 Two control room ventilation radiation monitoring channels shall be operable whenever the reactor coolant system is ~~above the cold shutdown condition~~ in MODES 1, 2, 3, or 4 or during ~~handling~~ movement of irradiated fuel. assemblies — A1

3.5.1.14 ~~The Main Steam Line Radiation Monitoring Instrumentation shall be operable with a minimum measurement range from 10^{-1} to 10^{4} mR/hr, whenever the reactor is above the cold shutdown condition.~~ — LA2 ODCM SAR

<LATER> (3.3C)

3.5.1.15 ~~Initiate functions of the EFIC system which are bypassed at cold shutdown conditions shall have the following minimum operability conditions:~~

~~a. "low steam generator pressure" initiate shall be operable when the main steam pressure exceeds 750 psig.~~

~~b. "loss of 4 RC pumps" initiate shall be operable when neutron flux exceeds 10% power.~~

~~c. "main feedwater pumps tripped" initiate shall be operable when neutron flux exceeds 10% power.~~ — LATER

<LATER> (3.3c + 3.7)

3.5.1.16 ~~The automatic steam generator isolation system within EFIC shall be operable when main steam pressure is greater than 750 psig.~~ — LATER

<Add 3.3.16 Condition C> — M4
<Add 3.3.16 Condition D>
<Add 3.3.16.2 Note> — L13

3.7.2
3.7.3

<LATER> (3.3D) 3.5.1.13 ~~Two control room ventilation radiation monitoring channels shall be operable whenever the reactor coolant system is above the cold shutdown condition or during handling of irradiated fuel.~~ LATER

3.5.1.14 ~~The Main Steam Line Radiation Monitoring Instrumentation shall be operable with a minimum measurement range from 10^{-1} to 10^{4} mR/hr, whenever the reactor is above the cold shutdown condition.~~ (R) TRM

<LATER> (3.3C) 3.5.1.15 ~~Initiate functions of the EFIC system which are bypassed at cold shutdown conditions shall have the following minimum operability conditions:~~ LATER

~~a. "low steam generator pressure" initiate shall be operable when the main steam pressure exceeds 750 psig.~~

~~b. "loss of 4 RC pumps" initiate shall be operable when neutron flux exceeds 10% power.~~

~~c. "main feedwater pumps tripped" initiate shall be operable when neutron flux exceeds 10% power.~~

SR 3.7.2.2 Note 2
SR 3.7.3.2 Note 2
+<LATER> (3.3C) 3.5.1.16 The automatic steam generator isolation system within EFIC shall be operable when main steam pressure is greater than 750 psig. +LATER

3.3.1
3.3.4
3.3.10

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless the requirements of Table 3.5.1-1, Columns 3 and 4, are met.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column 4 (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR, Section 7.

There are four reactor protection channels. Normal trip logic is two-out-of-four. Required trip logic for the power range instrumentation channels is two-out-of-three. Minimum trip logic on other instrumentation channels is one-out-of-two.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided with alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system channel bypass switch key permitted in the control room. Upon the discovery of inoperable functions in any one reactor protection channel, the effect of the failure on the reactor protection syst and other interconnected systems is evaluated. The affected reactor protect channel may be placed in channel bypass, remain in operation in a degraded condition, or placed in the tripped condition as determined by operating conditions and management judgment. This action allows placing the plant in the safest condition possible considering the extent of the failure, plant conditions, and guidance from plant management. Should the failure in the reactor protection channel prohibit the proper operation of another system, the appropriate actions for the affected system are implemented. Administra controls are established to preclude placing a reactor protection channel in channel bypass when any other reactor protection channel contains an inopera function in the untripped state. A2

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used. R TRM

The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps on the intermediate range scale.

The ESAS employs three independent and identical analog channels, which supply trip signals to two independent, identical digital subsystems. In order to actuate the safeguards systems, two out of three analog channels must trip. This will cause both digital subsystems to trip. Tripping of either digital subsystem will actuate all safeguards systems associated with that digital subsystem. A2

Because only one digital subsystem is necessary to actuate the safeguards systems and these systems are capable of tripping even when they are being tested, a single failure in a digital subsystem cannot prevent protective action.

3.3.5
3.3.6
3.3.7

A2

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless the requirements of Table 3.5.1-1, Columns 3 and 4, are met.

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There are four reactor protection channels. Normal trip logic is two-out-of-four. Required trip logic for the power range instrumentation channels is two-out-of-three. Minimum trip logic on other instrumentation channels is one-out-of-two.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided with alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system channel bypass switch key permitted in the control room. Upon the discovery of inoperable functions in any one reactor protection channel, the effect of the failure on the reactor protection syst and other interconnected systems is evaluated. The affected reactor protect channel may be placed in channel bypass, remain in operation in a degraded condition, or placed in the tripped condition as determined by operating conditions and management judgment. This action allows placing the plant in the safest condition possible considering the extent of the failure, plant conditions, and guidance from plant management. Should the failure in the reactor protection channel prohibit the proper operation of another system, the appropriate actions for the affected system are implemented. Administra controls are established to preclude placing a reactor protection channel in channel bypass when any other reactor protection channel contains an inopera function in the untripped state.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used.

R
TRM

The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps c the intermediate range scale.

A2

The ESAS employs three independent and identical analog channels, which supply trip signals to two independent, identical digital subsystems. In order to actuate the safeguards systems, two out of three analog channels must trip. This will cause both digital subsystems to trip. Tripping of either digital subsystem will actuate all safeguards systems associated with that digital subsystem.

Because only one digital subsystem is necessary to actuate the safeguards systems and these systems are capable of tripping even when they are being tested, a single failure in a digital subsystem cannot prevent protective action.

3.3.11
3.3.12
3.3.13

A2

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless the requirements of Table 3.5.1-1, Columns 3 and 4, are met.

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The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided with alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system channel bypass switch key permitted in the control room. Upon the discovery of inoperable functions in any one reactor protection channel, the effect of the failure on the reactor protection syst and other interconnected systems is evaluated. The affected reactor protect channel may be placed in channel bypass, remain in operation in a degraded condition, or placed in the tripped condition as determined by operating conditions and management judgment. This action allows placing the plant in the safest condition possible considering the extent of the failure, plant conditions, and guidance from plant management. Should the failure in the reactor protection channel prohibit the proper operation of another system, the appropriate actions for the affected system are implemented. Administra controls are established to preclude placing a reactor protection channel in channel bypass when any other reactor protection channel contains an inopera function in the untripped state.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used.

R
TRM

The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps on the intermediate range scale.

A2

The ESAS employs three independent and identical analog channels, which supply trip signals to two independent, identical digital subsystems. In order to actuate the safeguards systems, two out of three analog channels must trip. This will cause both digital subsystems to trip. Tripping of either digital subsystem will actuate all safeguards systems associated with that digital subsystem.

Because only one digital subsystem is necessary to actuate the safeguards systems and these systems are capable of tripping even when they are being tested, a single failure in a digital subsystem cannot prevent protective action.

3.3.8
3.3.15
3.3.16

(A2)

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless the requirements of Table 3.5.1-1, Columns 3 and 4, are met.

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The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided with alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system channel bypass switch key permitted in the control room. Upon the discovery of inoperable functions in any one reactor protection channel, the effect of the failure on the reactor protection syst and other interconnected systems is evaluated. The affected reactor protect channel may be placed in channel bypass, remain in operation in a degraded condition, or placed in the tripped condition as determined by operating conditions and management judgment. This action allows placing the plant in the safest condition possible considering the extent of the failure, plant conditions, and guidance from plant management. Should the failure in the reactor protection channel prohibit the proper operation of another system, the appropriate actions for the affected system are implemented. Administra controls are established to preclude placing a reactor protection channel in channel bypass when any other reactor protection channel contains an inopera function in the untripped state.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used.

(R) TRM

The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps c the intermediate range scale.

The ESAS employs three independent and identical analog channels, which supply trip signals to two independent, identical digital subsystems. In order to actuate the safeguards systems, two out of three analog channels must trip. This will cause both digital subsystems to trip. Tripping of either digital subsystem will actuate all safeguards systems associated with that digital subsystem.

(A2)

Because only one digital subsystem is necessary to actuate the safeguards systems and these systems are capable of tripping even when they are being tested, a single failure in a digital subsystem cannot prevent protective action.

Items on this page also addressed in the following packages: 3.3 A, 3.3 B, 3.3 C, 3.3 D

SR

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless the requirements of Table 3.5.1-1, Columns 3 and 4, are met.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column 4 (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR, Section 7.

There are four reactor protection channels. Normal trip logic is two-out-of-four. Required trip logic for the power range instrumentation channels is two-out-of-three. Minimum trip logic on other instrumentation channels is one-out-of-two.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided with alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system channel bypass switch key permitted in the control room. Upon the discovery of inoperable functions in any one reactor protection channel, the effect of the failure on the reactor protection system and other interconnected systems is evaluated. The affected reactor protection channel may be placed in channel bypass, remain in operation in a degraded condition, or placed in the tripped condition as determined by operating conditions and management judgment. This action allows placing the plant in the safest condition possible considering the extent of the failure, plant conditions, and guidance from plant management. Should the failure in the reactor protection channel prohibit the proper operation of another system, the appropriate actions for the affected system are implemented. Administrative controls are established to preclude placing a reactor protection channel in channel bypass when any other reactor protection channel contains an inoperable function in the untripped state.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used. R TRM

The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps on the intermediate range scale.

The ESAS employs three independent and identical analog channels, which supply trip signals to two independent, identical digital subsystems. In order to actuate the safeguards systems, two out of three analog channels must trip. This will cause both digital subsystems to trip. Tripping of either digital subsystem will actuate all safeguards systems associated with that digital subsystem.

Because only one digital subsystem is necessary to actuate the safeguards systems and these systems are capable of tripping even when they are being tested, a single failure in a digital subsystem cannot prevent protective action.

3.3.1
3.3.4
3.3.10

Removal of a module required for protection from a RPS channel will cause that channel to trip, unless that channel has been bypassed, so that only one channel of the other three must trip to cause a reactor trip. Thus, sufficient redundancy has been built into the system to cover this situation.

Removal of a module required for protective action from an analog ESAS channel will cause that channel to trip, so that only one of the other two must trip to actuate the safeguards systems. Removal of a module required for protective action from a digital ESAS subsystem will not cause that subsystem to trip. The fact that a module has been removed will be continuously annunciated to the operator. The redundant digital subsystem is still sufficient to indicate complete ESAS action.

The testing schemes of the RPS, the ESAS, and the EFIC enables complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.

The EFIC is designed to allow testing during power operation. One channel may be placed in key locked "maintenance bypass" prior to testing. This will bypass only one channel of EFW initiate logic. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC receives signals from the NI/RPS, the maintenance bypass from the NI/RPS is interlocked with the EFIC. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. Prior to placing a channel of EFIC in maintenance bypass, any NI/RPS channel containing inoperable functions in the untripped state is evaluated for its effect on EFIC. Only the EFIC channel corresponding to the NI/RPS channel containing the inoperable function may be placed in maintenance bypass unles it can be shown that the failure in the NI/RPS channel has no effect on EFIC actuation, actions are taken to ensure EFIC actuation when required, or the appropriate actions of Table 3.5.1-1 are implemented. The EFIC can be teste from its input terminals to the actuated device controllers. A test of the EFIC trip logic will actuate one of two relays in the controllers. Activati of both relays is required in order to actuate the controllers. The two relays are tested individually to prevent automatic actuation of the compone The EFIC trip logic is two (one-out-of-two).

Reactor trips on loss of all main feedwater and on turbine trips will sense the start of a loss of OTSG heat sink and actuate earlier than other trip signals. This early actuation will provide a lower peak RC pressure during the initial over pressurization following a loss of feedwater or turbine trip event. The LOFW trip may be bypassed up to 10% to allow sufficient margin for bringing the MFW pumps into use at approximately 7%. The Turbine Trip may be bypassed up to 45% based on BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985 and the NRC Safety Evaluation Report for BAW-1893 issued from Mr. D. M. Crutchfield to Mr. J. H. Taylor via letter dated April 25, 1986.

The Automatic Closure and Isolation System (ACI) is designed to close the Decay Heat Removal System (DHRS) return line isolation valves when the Reactor Coolant System (RCS) pressure exceeds a selected fraction of the DHRS design pressure or when core flooding system isolation valves are opened. The ACI is designed to permit manual operation of the DHRS return line isolation valves when permissive conditions exist. In addition, the ACI is designed to disallow manual operation of the valves when permissive conditions do not exist.

A2

3.3.5
3.3.6
3.3.7

A2

Removal of a module required for protection from a RPS channel will cause that channel to trip, unless that channel has been bypassed, so that only one channel of the other three must trip to cause a reactor trip. Thus, sufficient redundancy has been built into the system to cover this situation.

Removal of a module required for protective action from an analog ESAS channel will cause that channel to trip, so that only one of the other two must trip to actuate the safeguards systems. Removal of a module required for protective action from a digital ESAS subsystem will not cause that subsystem to trip. The fact that a module has been removed will be continuously annunciated to the operator. The redundant digital subsystem is still sufficient to indicate complete ESAS action.

The testing schemes of the RPS, the ESAS, and the EFIC enables complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.

The EFIC is designed to allow testing during power operation. One channel may be placed in key locked "maintenance bypass" prior to testing. This will bypass only one channel of EFW initiate logic. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC receives signals from the NI/RPS, the maintenance bypass from the NI/RPS is interlocked with the EFIC. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. Prior to placing a channel of EFIC in maintenance bypass, any NI/RPS channel containing inoperable functions in the untripped state is evaluated for its effect on EFIC. Only the EFIC channel corresponding to the NI/RPS channel containing the inoperable function may be placed in maintenance bypass unles it can be shown that the failure in the NI/RPS channel has no effect on EFIC actuation, actions are taken to ensure EFIC actuation when required, or the appropriate actions of Table 3.5.1-1 are implemented. The EFIC can be teste from its input terminals to the actuated device controllers. A test of the EFIC trip logic will actuate one of two relays in the controllers. Activati of both relays is required in order to actuate the controllers. The two relays are tested individually to prevent automatic actuation of the compone The EFIC trip logic is two (one-out-of-two).

Reactor trips on loss of all main feedwater and on turbine trips will sense the start of a loss of OTSG heat sink and actuate earlier than other trip signals. This early actuation will provide a lower peak RC pressure during the initial over pressurization following a loss of feedwater or turbine trip event. The LOFW trip may be bypassed up to 10% to allow sufficient margin for bringing the MFW pumps into use at approximately 7%. The Turbine Trip may be bypassed up to 45% based on BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985 and the NRC Safety Evaluation Report for BAW-1893 issued from Mr. D. M. Crutchfield to Mr. J. H. Taylor via letter dated April 25, 1986.

The Automatic Closure and Isolation System (ACI) is designed to close the Decay Heat Removal System (DHRS) return line isolation valves when the Reactor Coolant System (RCS) pressure exceeds a selected fraction of the DHRS design pressure or when core flooding system isolation valves are opened. The ACI is designed to permit manual operation of the DHRS return line isolation valves when permissive conditions exist. In addition, the ACI is designed to disallow manual operation of the valves when permissive conditions do not exist.

3.3.11
3.3.12
3.3.13
A2

Removal of a module required for protection from a RPS channel will cause that channel to trip, unless that channel has been bypassed, so that only one channel of the other three must trip to cause a reactor trip. Thus, sufficient redundancy has been built into the system to cover this situation.

Removal of a module required for protective action from an analog ESAS channel will cause that channel to trip, so that only one of the other two must trip to actuate the safeguards systems. Removal of a module required for protective action from a digital ESAS subsystem will not cause that subsystem to trip. The fact that a module has been removed will be continuously annunciated to the operator. The redundant digital subsystem is still sufficient to indicate complete ESAS action.

The testing schemes of the RPS, the ESAS, and the EFIC enables complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.

The EFIC is designed to allow testing during power operation. One channel may be placed in key locked "maintenance bypass" prior to testing. This will bypass only one channel of EFW initiate logic. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC receives signals from the NI/RPS, the maintenance bypass from the NI/RPS is interlocked with the EFIC. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. Prior to placing a channel of EFIC in maintenance bypass, any NI/RPS channel containing inoperable functions in the untripped state is evaluated for its effect on EFIC. Only the EFIC channel corresponding to the NI/RPS channel containing the inoperable function may be placed in maintenance bypass unles it can be shown that the failure in the NI/RPS channel has no effect on EFIC actuation, actions are taken to ensure EFIC actuation when required, or the appropriate actions of Table 3.5.1-1 are implemented. The EFIC can be teste from its input terminals to the actuated device controllers. A test of the EFIC trip logic will actuate one of two relays in the controllers. Activat: of both relays is required in order to actuate the controllers. The two relays are tested individually to prevent automatic actuation of the compone The EFIC trip logic is two (one-out-of-two).

Reactor trips on loss of all main feedwater and on turbine trips will sense the start of a loss of OTSG heat sink and actuate earlier than other trip signals. This early actuation will provide a lower peak RC pressure during the initial over pressurization following a loss of feedwater or turbine trip event. The LOFW trip may be bypassed up to 10% to allow sufficient margin for bringing the MFW pumps into use at approximately 7%. The Turbine Trip may be bypassed up to 45% based on BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985 and the NRC Safety Evaluation Report for BAW-1893 issued from Mr. D. M. Crutchfield to Mr. J. H. Taylor via letter dated April 25, 1986.

The Automatic Closure and Isolation System (ACI) is designed to close the Decay Heat Removal System (DHRS) return line isolation valves when the Reactor Coolant System (RCS) pressure exceeds a selected fraction of the DHRS design pressure or when core flooding system isolation valves are opened. The ACI is designed to permit manual operation of the DHRS return line isolation valves when permissive conditions exist. In addition, the ACI is designed to disallow manual operation of the valves when permissive conditions do not exist.

3.3.8
3.3.15
3.3.16

A2

Removal of a module required for protection from a RPS channel will cause that channel to trip, unless that channel has been bypassed, so that only one channel of the other three must trip to cause a reactor trip. Thus, sufficient redundancy has been built into the system to cover this situation.

Removal of a module required for protective action from an analog ESAS channel will cause that channel to trip, so that only one of the other two must trip to actuate the safeguards systems. Removal of a module required for protective action from a digital ESAS subsystem will not cause that subsystem to trip. The fact that a module has been removed will be continuously annunciated to the operator. The redundant digital subsystem is still sufficient to indicate complete ESAS action.

The testing schemes of the RPS, the ESAS, and the EFIC enables complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.

The EFIC is designed to allow testing during power operation. One channel may be placed in key locked "maintenance bypass" prior to testing. This will bypass only one channel of EFW initiate logic. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC receives signals from the NI/RPS, the maintenance bypass from the NI/RPS is interlocked with the EFIC. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. Prior to placing a channel of EFIC in maintenance bypass, any NI/RPS channel containing inoperable functions in the untripped state is evaluated for its effect on EFIC. Only the EFIC channel corresponding to the NI/RPS channel containing the inoperable function may be placed in maintenance bypass unles it can be shown that the failure in the NI/RPS channel has no effect on EFIC actuation, actions are taken to ensure EFIC actuation when required, or the appropriate actions of Table 3.5.1-1 are implemented. The EFIC can be teste from its input terminals to the actuated device controllers. A test of the EFIC trip logic will actuate one of two relays in the controllers. Activat: of both relays is required in order to actuate the controllers. The two relays are tested individually to prevent automatic actuation of the compone The EFIC trip logic is two (one-out-of-two).

Reactor trips on loss of all main feedwater and on turbine trips will sense the start of a loss of OTSG heat sink and actuate earlier than other trip signals. This early actuation will provide a lower peak RC pressure during the initial over pressurization following a loss of feedwater or turbine trip event. The LOFW trip may be bypassed up to 10% to allow sufficient margin for bringing the MFW pumps into use at approximately 7%. The Turbine Trip may be bypassed up to 45% based on BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985 and the NRC Safety Evaluation Report for BAW-1893 issued from Mr. D. M. Crutchfield to Mr. J. H. Taylor via letter dated April 25, 1986.

The Automatic Closure and Isolation System (ACI) is designed to close the Decay Heat Removal System (DHRS) return line isolation valves when the Reactor Coolant System (RCS) pressure exceeds a selected fraction of the DHRS design pressure or when core flooding system isolation valves are opened. The ACI is designed to permit manual operation of the DHRS return line isolation valves when permissive conditions exist. In addition, the ACI is designed to disallow manual operation of the valves when permissive conditions do not exist.

3.4.14

A2

Removal of a module required for protection from a RPS channel will cause that channel to trip, unless that channel has been bypassed, so that only one channel of the other three must trip to cause a reactor trip. Thus, sufficient redundancy has been built into the system to cover this situation.

Removal of a module required for protective action from an analog ESAS channel will cause that channel to trip, so that only one of the other two must trip to actuate the safeguards systems. Removal of a module required for protective action from a digital ESAS subsystem will not cause that subsystem to trip. The fact that a module has been removed will be continuously annunciated to the operator. The redundant digital subsystem is still sufficient to indicate complete ESAS action.

The testing schemes of the RPS, the ESAS, and the EFIC enables complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.

The EFIC is designed to allow testing during power operation. One channel may be placed in key locked "maintenance bypass" prior to testing. This will bypass only one channel of EFW initiate logic. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC receives signals from the NI/RPS, the maintenance bypass from the NI/RPS is interlocked with the EFIC. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. Prior to placing a channel of EFIC in maintenance bypass, any NI/RPS channel containing inoperable functions in the untripped state is evaluated for its effect on EFIC. Only the EFIC channel corresponding to the NI/RPS channel containing the inoperable function may be placed in maintenance bypass unles it can be shown that the failure in the NI/RPS channel has no effect on EFIC actuation, actions are taken to ensure EFIC actuation when required, or the appropriate actions of Table 3.5.1-1 are implemented. The EFIC can be teste from its input terminals to the actuated device controllers. A test of the EFIC trip logic will actuate one of two relays in the controllers. Activati of both relays is required in order to actuate the controllers. The two relays are tested individually to prevent automatic actuation of the compone The EFIC trip logic is two (one-out-of-two).

Reactor trips on loss of all main feedwater and on turbine trips will sense the start of a loss of OTSG heat sink and actuate earlier than other trip signals. This early actuation will provide a lower peak RC pressure during the initial over pressurization following a loss of feedwater or turbine trip event. The LOFW trip may be bypassed up to 10% to allow sufficient margin for bringing the MFW pumps into use at approximately 7%. The Turbine Trip may be bypassed up to 45% based on BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985 and the NRC Safety Evaluation Report for BAW-1893 issued from Mr. D. M. Crutchfield to Mr. J. H. Taylor via letter dated April 25, 1986.

The Automatic Closure and Isolation System (ACI) is designed to close the Decay Heat Removal System (DHRS) return line isolation valves when the Reactor Coolant System (RCS) pressure exceeds a selected fraction of the DHRS design pressure or when core flooding system isolation valves are opened. The ACI is designed to permit manual operation of the DHRS return line isolation valves when permissive conditions exist. In addition, the ACI is designed to disallow manual operation of the valves when permissive conditions do not exist.

3.3.1
3.3.4
3.3.10

Power is normally supplied to the control rod drive mechanisms from two separate parallel 480 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state, on-line repairs to the failed device, when practical, will be made and the remaining trip devices will be tested. Four hours is ample time to test the remaining trip devices and, in many cases, make on-line repairs.

The Degraded Voltage Monitoring relay settings are based on the short term starting voltage protection as well as long term running voltage protection. The 4.16 KV undervoltage relay setpoints are based on the allowable starting voltage plus maximum system voltage drops to the motor terminals, which allows approximately 78% of motor rated voltage at the motor terminals. The 460V undervoltage relay setpoint is based on long term motor voltage requirements plus the maximum feeder voltage drop allowance resulting in a 92% setting of motor rated voltage.

A2

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendation of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

The subcooled margin monitors (SMM), and core-exit thermocouples (CET), Reactor Vessel Level Monitoring System (RVLMS) and Hot Leg Level Measurement System (HLLMS) are a result of the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737. The function of the ICC instrumentation is to increase the ability of the plant operators to diagnose the approach to and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37 and are not required by the accident analysis, nor to bring the plant to cold shutdown conditions. The Reactor Vessel Level Monitor is provided as a means of indicating level in the reactor vessel during accident conditions. The channel operability of the RVLMS is defined as a minimum of three sensors in the upper plenum region and two sensors in the dome region operable. When Reactor Coolant Pumps are running, all except the dome sensors are interlocked to read "invalid" due to flow induced variables that may offset the sensor outputs. The channel operability of the HLLMS is defined as a minimum of one wide range and any two of the narrow range transmitters in the same channel operable. If the equipment is inaccessible due to health and industrial safety concerns (for example, high radiation area, low oxygen content of the containment atmosphere) or due to physical location of the fault (for example, probe failure in the reactor vessel), then operation may continue until the next scheduled refueling outage and a report filed.

3.3.11
3.3.12
3.3.13

A2

Power is normally supplied to the control rod drive mechanisms from two separate parallel 480 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state, on-line repairs to the failed device, when practical, will be made and the remaining trip devices will be tested. Four hours is ample time to test the remaining trip devices and, in many cases, make on-line repairs.

The Degraded Voltage Monitoring relay settings are based on the short term starting voltage protection as well as long term running voltage protection. The 4.16 KV undervoltage relay setpoints are based on the allowable starting voltage plus maximum system voltage drops to the motor terminals, which allows approximately 78% of motor rated voltage at the motor terminals. The 460V undervoltage relay setpoint is based on long term motor voltage requirements plus the maximum feeder voltage drop allowance resulting in a 92% setting of motor rated voltage.

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendation of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

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3.3.8
3.3.15
3.3.16

A2

Power is normally supplied to the control rod drive mechanisms from two separate parallel 480 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state, on-line repairs to the failed device, when practical, will be made and the remaining trip devices will be tested. Four hours is ample time to test the remaining trip devices and, in many cases, make on-line repairs.

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The subcooled margin monitors (SMM), and core-exit thermocouples (CET), Reactor Vessel Level Monitoring System (RVLMS) and Hot Leg Level Measurement System (HLLMS) are a result of the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737. The function of the ICC instrumentation is to increase the ability of the plant operators to diagnose the approach to and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37 and are not required by the accident analysis, nor to bring the plant to cold shutdown conditions. The Reactor Vessel Level Monitor is provided as a means of indicating level in the reactor vessel during accident conditions. The channel operability of the RVLMS is defined as a minimum of three sensors in the upper plenum region and two sensors in the dome region operable. When Reactor Coolant Pumps are running, all except the dome sensors are interlocked to read "invalid" due to flow induced variables that may offset the sensor outputs. The channel operability of the HLLMS is defined as a minimum of one wide range and any two of the narrow range transmitters in the same channel operable. If the equipment is inaccessible due to health and industrial safety concerns (for example, high radiation area, low oxygen content of the containment atmosphere) or due to physical location of the fault (for example, probe failure in the reactor vessel), then operation may continue until the next scheduled refueling outage and a report filed.

3.3.1
3.3.4
3.3.10

The principal function of the Control Room Isolation-High Radiation is to provide an enclosed environment from which the unit can be operated following an uncontrolled release of radioactivity. Due to the unique arrangement of the shared control room envelope, one control room isolation channel receives a high radiation signal from the ANO-1 control room ventilation intake duct monitor and the redundant channel receives a high radiation signal from the ANO-2 control room ventilation intake duct monitor. With no channel of the control room radiation monitoring system operable, the CREVS must be placed in a condition that does not require the isolation to occur (i.e., one operable train of CREVS is placed in the emergency recirculation mode of operation). Reactor operation may continue indefinitely in this state.

To support loss of main feedwater analyses, steam line/feedwater line break analyses, SBLOCA analyses, and NUREG-0737 requirements, the EFIC system is designed to automatically initiate EFW when:

1. all four RC pumps are tripped
2. both main feedwater pumps are tripped
3. the level of either steam generator is low
4. either steam generator pressure is low
5. ESAS ECCS actuation (high RB pressure or low RCS pressure)

The EFIC system is also designed to isolate the affected steam generator on a steam line/feedwater line break and supply EFW to the intact generator according to the following logic: A2

- If both SG's are above 600 psig, supply EFW to both SG's.
- If one SG is below 600 psig, supply EFW to the other SG.
- If both SG's are below 600 psig, but the pressure difference between the two SG's exceeds 100 psig, supply EFW only to the SG with the higher pressure.
- If both SG's are below 600 psig and the pressure difference is less than 100 psig, supply EFW to both SG's.

At cold shutdown conditions all EFIC initiate and isolate functions are bypassed except low steam generator level initiate. The bypassed functions will be automatically reset at the values or plant conditions identified in Specification 3.5.1.15. "Loss of 4 RC pumps" initiate and "low steam generator pressure" initiate are the only shutdown bypasses to be manually initiated during cooldown. If reset is not done manually, they will automatically reset. Main feedwater pump trip bypass is automatically removed above 10% power.

REFERENCE

FSAR, Section 7.1

3.3.11
3.3.12
3.3.13

~~The principal function of the Control Room Isolation-High Radiation is to provide an enclosed environment from which the unit can be operated following an uncontrolled release of radioactivity. Due to the unique arrangement of the shared control room envelope, one control room isolation channel receives a high radiation signal from the ANO-1 control room ventilation intake duct monitor and the redundant channel receives a high radiation signal from the ANO-2 control room ventilation intake duct monitor. With no channel of the control room radiation monitoring system operable, the CREVS must be placed in a condition that does not require the isolation to occur (i.e., one operable train of CREVS is placed in the emergency recirculation mode of operation). Reactor operation may continue indefinitely in this state.~~

~~To support loss of main feedwater analyses, steam line/feedwater line break analyses, SBLOCA analyses, and NUREG-0737 requirements, the EFIC system is designed to automatically initiate EFW when:~~

1. ~~all four RC pumps are tripped~~
2. ~~both main feedwater pumps are tripped~~
3. ~~the level of either steam generator is low~~
4. ~~either steam generator pressure is low~~
5. ~~ESAS ECCS actuation (high RB pressure or low RCS pressure)~~

~~The EFIC system is also designed to isolate the affected steam generator on a steam line/feedwater line break and supply EFW to the intact generator according to the following logic:~~

- ~~If both SG's are above 600 psig, supply EFW to both SG's.~~
- ~~If one SG is below 600 psig, supply EFW to the other SG.~~
- ~~If both SG's are below 600 psig, but the pressure difference between the two SG's exceeds 100 psig, supply EFW only to the SG with the higher pressure.~~
- ~~If both SG's are below 600 psig and the pressure difference is less than 100 psig, supply EFW to both SG's.~~

~~At cold shutdown conditions all EFIC initiate and isolate functions are bypassed except low steam generator level initiate. The bypassed functions will be automatically reset at the values or plant conditions identified in Specification 3.5.1.15. "Loss of 4 RC pumps" initiate and "low steam generator pressure" initiate are the only shutdown bypasses to be manually initiated during cooldown. If reset is not done manually, they will automatically reset. Main feedwater pump trip bypass is automatically removed above 10% power.~~

~~**REFERENCE**~~

~~FSAR, Section 7.1~~

A2

3.3.8
3.3.15
3.3.16

The principal function of the Control Room Isolation-High Radiation is to provide an enclosed environment from which the unit can be operated following an uncontrolled release of radioactivity. Due to the unique arrangement of the shared control room envelope, one control room isolation channel receives a high radiation signal from the ANO-1 control room ventilation intake duct monitor and the redundant channel receives a high radiation signal from the ANO-2 control room ventilation intake duct monitor. With no channel of the control room radiation monitoring system operable, the CREVS must be placed in a condition that does not require the isolation to occur (i.e., one operable train of CREVS is placed in the emergency recirculation mode of operation). Reactor operation may continue indefinitely in this state.

To support loss of main feedwater analyses, steam line/feedwater line break analyses, SBLOCA analyses, and NUREG-0737 requirements, the EFIC system is designed to automatically initiate EFW when: (A2)

1. all four RC pumps are tripped
2. both main feedwater pumps are tripped
3. the level of either steam generator is low
4. either steam generator pressure is low
5. ESAS ECCS actuation (high RB pressure or low RCS pressure)

The EFIC system is also designed to isolate the affected steam generator on a steam line/feedwater line break and supply EFW to the intact generator according to the following logic:

- If both SG's are above 600 psig, supply EFW to both SG's.
- If one SG is below 600 psig, supply EFW to the other SG.
- If both SG's are below 600 psig, but the pressure difference between the two SG's exceeds 100 psig, supply EFW only to the SG with the higher pressure.
- If both SG's are below 600 psig and the pressure difference is less than 100 psig, supply EFW to both SG's.

At cold shutdown conditions all EFIC initiate and isolate functions are bypassed except low steam generator level initiate. The bypassed functions will be automatically reset at the values or plant conditions identified in Specification 3.5.1.15. "Loss of 4 RC pumps" initiate and "low steam generator pressure" initiate are the only shutdown bypasses to be manually initiated during cooldown. If reset is not done manually, they will automatically reset. Main feedwater pump trip bypass is automatically removed above 10% power.

REFERENCE

FSAR, Section 7.1

<Add 3.3.1 Appl.> A1

<Add 3.3.1 Condition C> M18

<Add 3.3.1 Condition E> M11

<Add 3.3.1 Surveillance Requirements - NOTE> A1

LA1 BASES

A3

A7

3.3.1 LCO — **Table 3.5.1-1 Instrumentation Limiting Conditions for Operation (Note 6)**

REACTOR PROTECTION SYSTEM

A6

	Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
3.3.2 LCO	1. Manual pushbutton	1	1	1	0	Note 1
T3.3.1-1, #1a	2. ~~Power range~~ nuclear overpower instrument channel	4	2	~~3(Note 4)~~ 2	~~1(Note 4)~~ 0	Note 1
3.3.10 LCO	3. Intermediate range instrument channels	2	Note 7	1	0	Notes ~~1~~, 2
3.3.9 LCO	4. Source range instrument channels	2	Note 7	1	0	Notes ~~1~~, 2, 3
T3.3.1-1, #2	5. Reactor coolant temperature instrument channels	4	2	2	1	Note 1
T3.3.1-1, #5 &	6. Pressure-temperature instrument channels	4	2	2	1	Note 1
T3.3.1-1, #8	7. Flux/imbalance/flow instrument channels	4	2	2	1	Note 1
	8. Reactor coolant pressure					
T3.3.1-1, #3	a. High reactor coolant pressure instrument channels	4	2	2	1	Note 1
T3.3.1-1, #4	b. Low reactor coolant pressure instrument channels	4	2	2	1	Note 1
T3.3.1-1, #7	9. Power/number of pumps instrument channels	4	2	2	1	Note 1
T3.3.1-1, #6	10. High reactor building pressure channels	4	2	2	1	Note 1

A3 M1 L5 A1 M8 A1

<Add Table 3.3.1-1 APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS for Functions 1 thru 8 & 11, including T3.3.1-1 Notes a, b, c & d.> M12

3.3.1
3.3.2
3.3.9
3.3.10

Table 3.5.1-1 Instrumentation Limiting Conditions for Operation

(Note 6)

(A1)

REACTOR PROTECTION SYSTEM

	1	2	3	4	5
Functional Unit	No. of channels	No. of channels for system trip	Min. operable channels	Min. degree of redundancy	Operator action if conditions of column 3 or 4 cannot be met
1. Manual pushbutton	1	1	1	0	Note 1
2. Power range instrument channel	4	2	3(Note 4)	1(Note 4)	Note 1
3. Intermediate range instrument channels	2	Note 7	1	0	Notes 1, 2
4. Source range instrument channels	2	Note 7	1	0	Notes 1, 2, 3
5. Reactor coolant temperature instrument channels	4	2	2	1	Note 1
6. Pressure-temperature instrument channels	4	2	2	1	Note 1
7. Flux/imbalance/flow instrument channels	4	2	2	1	Note 1
8. Reactor coolant pressure					
a. High reactor coolant pressure instrument channels	4	2	2	1	Note 1
b. Low reactor coolant pressure instrument channels	4	2	2	1	Note 1
9. Power/number of pumps instrument channels	4	2	2	1	Note 1
10. High reactor building pressure channels	4	2	2	1	Note 1

(LATER) (3.3A)

LATER

3.3.11

<Add Table 3.3.1-1 Function 1b, Nuclear Overpower Low Setpoint, Condition & SRs> M11

<Add Table 3.3.1-1 Function 11, Shutdown Bypass RCS High Pressure, Condition & SRs> M11

LA1 BASES

A3

A7

Table 3.5.1-1 (Cont'd)

Amendment No. 67, 117

REACTOR PROTECTION SYSTEM (Cont'd)

A3

Margin mark	Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met	Mark
T3.3.1-1, function 10	11. Reactor trip upon loss of Main Feedwater	4	2	2	1	Notes 1, 15	A1
function 9	12. Reactor trip upon turbine trip	4	2	2	1	Notes 1, 16	A1
3.3.4 LCO c	13. Electronic (SCR) Trip Relay	2	2	2	0	Note 23	A8
	14. Control Rod Drive Trip Breakers						
3.3.4 LCO a	A. AC Breakers	2	2	2	0	Notes 24, 25	A1
3.3.4 LCO b	B. DC Breakers ~~(Note 26)~~	2	2	2	0	Notes 24, 25	A1

4#a

LA1 BASES

<Add 3.3.2 Appl, Conditions A & C> M1

<Add 3.3.3 LCO, Appl, and ACTIONS> M14

<Add 3.3.4 ACTIONS - NOTE> A9

<Add 3.3.4 Appl, Conditions D & E> M15

<Add 3.3.9 Appl, Condition A> M16

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9

Add 3.3.5 ACTIONS NOTE
Add 3.3.6 ACTIONS NOTE
Add 3.3.7 ACTIONS NOTE

A5 LA1 Bases A3 A7

Table 3.5.1-1 (Cont'd)

ENGINEERED SAFEGUARDS ACTUATION SYSTEM

	Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met	
3.3.7 LCO	1. Highpressure injection system (Note 8) (Actuation Logic channels)						A9
3.3.5 LCO Table 3.3.5-1 parameter 1. & LATER (3.3D)	a. Reactor coolant pressure instrument channels	3	2	3 (Note 6)	1	Notes 1, 5	A6 & LATER
Parameter 2.	b. Reactor building 4 psig instrument channels	3	2	3 (Note 6)	1	Notes 1, 5	A6
3.3.6 LCO *a.	c. Manual trip pushbutton	2	1	2	1	Notes 1, 5	A8
3.3.7 LCO 45	2. Low pressure injection system (Note 8) (Actuation Logic Channels)						A9
3.3.5 LCO Table 3.3.5-1 Parameter 1. & LATER (3.3D)	a. Reactor coolant pressure instrument channels	3	2	3 (Note 6)	1	Notes 1, 5	A6 & LATER
Parameter 2.	b. Reactor building 4 psig instrument channels	3	2	3 (Note 6)	1	Notes 1, 5	A6
3.3.6 LCO *b.	c. Manual trip pushbutton	2	1	2	1	Notes 1, 5	A8
3.3.7 LCO	3. Reactor building isolation and reactor building cooling system (Note 8) (Actuation Logic Channels)						A9
3.3.5 LCO Table 3.3.5-1 Parameter 2.	a. Reactor building 4 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5	A6
3.3.6 LCO *c.	b. Manual trip pushbutton	2	1	2	1	Notes 1, 5	A8

<Add 3.3.5 RA B.2.2 Note>
<Add 3.3.5 Appl. & Table 3.3.5-1 Applicable MODES or Other Specified Condition> L1
<Add 3.3.5 RA B.2.1 with NOTE>

3.3.5
3.3.6
3.3.7

Table 3.5.1-1 (Cont'd)

ENGINEERED SAFEGUARDS ACTUATION SYSTEM

LA1 Bases

A3

A7

Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
1. High pressure injection system (Note 8)					
a. Reactor coolant pressure instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
b. Reactor building 4 psig instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
c. Manual trip pushbutton	2	1	2	1	Notes 1, 5
2. Low pressure injection system (Note 8)					
a. Reactor coolant pressure instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
b. Reactor building 4 psig instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
c. Manual trip pushbutton	2	1	2	1	Notes 1, 5
3. Reactor building isolation and reactor building cooling system (Note 8)					
a. Reactor building 4 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5
b. Manual trip pushbutton	2	1	2	1	Notes 1, 5

M7 & LATER

LATER

Table 3.3.15-1 PAM #4 <LATER> (3.3B)

<LATER> (3.3B)

45

LA1 Bases

A3

A7

M7 & LATER

Table 3.3.15-1 #PAM #4 <Later> (3.3B)

LATER

<LATER> (3.3B)

3.3.15

Amendment No. 91

Table 3.5.1-1 (Cont'd

ENGINEERED SAFEGUARDS ACTUATION SYSTEM
(Cont'd)

LA1 Bases

A3

A7

	Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met	
3.3.7 LCO	4. Reactor building spray pumps (Note 8) Actuation Logic Channels						A9
3.3.5 LCO Table 3.3.5-1 Parameter 3.	a. Reactor building 30 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5	A6
3.3.6 LCO d.	b. Manual trip pushbutton	2	1	2	1	Notes 1, 5	A8
3.3.7 LCO	5. Reactor building spray valves (Note 8) Actuation Logic Channels						A9
3.3.5 LCO Table 3.3.5-1 Parameter 3.	a. Reactor building 30 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5	A6
3.3.6 LCO e.	b. Manual trip pushbutton	2	1	2	1	Notes 1, 5	A8

45a

EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM

<LATER> (3.3C)	1. EFW Initiation						
	a. Manual	2	1	2	1	Note 1	LATER

<ADD 3.3.6 Appl.>
<ADD 3.3.7 Appl.> L5

3.3.5
3.3.6
3.3.7

Add 3.3.11 ACTIONS Note
Add 3.3.12 ACTIONS Note
Add 3.3.13 ACTIONS Note
Add 3.3.11 SR Note

A5

LA1 Bases

A3

A7

Table 3.5.1-1 (Cont'd)

Amendment No. 91

ENGINEERED SAFEGUARDS ACTUATION SYSTEM (Cont'd)

Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
4. Reactor building spray pumps (Note 8)					
a. Reactor building 30 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5
b. Manual trip pushbutton	2	1	2	1	Notes 1, 5
5. Reactor building spray valves (Note 8)					
a. Reactor building 30 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5
b. Manual trip pushbutton	2	1	2	1	Notes 1, 5

<LATER> (3.3B)

LATER

15a

EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM

LA1 Bases

A3

A7

Functional Unit	1	2	3	4	5
1. EFW Initiation					
a. Manual	2	1	2	1	Note 1

3.3.12 LCO #C

A6

<Add 3.3.12 RA A.1 & B.1> L1

<Add 3.3.12 Appl. & RA C.2, D.2.1 & D.2.2> M6

3.3.11
3.3.12
3.3.13

Table 3.5.1-1 (Cont'd)

LA1 Bases

A3

A7

EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM (Cont'd)

T3.3.11-1

Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met	
#1.b b. Low Level SG A or B	~~4/SG~~	~~2/SG~~	~~2/SG~~	~~1~~	Note 1	A11
#1.c &LATER(3.3D) c. Low Pressure, SG A or B	~~4/SG~~	~~2/SG~~	~~2/SG~~	~~1~~	Note 1, 19	& LATER
#1.a d. Loss of Both MFW Pumps and PWR > 10%	~~4~~	~~2~~	~~2~~	~~1~~	Note 1	A11
#1.d e. Loss of 4 RC Pumps	~~4~~	~~2~~	~~2~~	~~1~~	Note 1, 15	
3.3.13 LCO#b f. ESAS Actuation Logic Tripped	~~2~~	~~1~~	~~2~~	~~1~~	Note 1	A12
2. SG-A Main Steam Line Isolation						
3.3.12 LCO#a a. Manual	~~2~~	~~1~~	~~2~~	~~1~~	Note 1	A6
T3.3.11-1, #3a b. Low SG A Pressure	~~4~~	~~2~~	~~2~~	~~1~~	Note 1, 19	A11
3. SG-B Main Steam Line Isolation						
3.3.12 LCO#b a. Manual	~~2~~	~~1~~	~~2~~	~~1~~	Note 1	A6
T3.3.11-1, #3.a b. Low SG B Pressure	~~4~~	~~2~~	~~2~~	~~1~~	Note 1, 19	A11

⟨Add 3.3.13 LCO#a with Appl., Cond. A, RAs C.2.1 & C.2.2, and SR 3.3.13.1⟩ M7

⟨Add 3.3.13 LCO#b Appl., RAs A.1 & B.2, and SR 3.3.13.1⟩ M7

⟨Add ITS 3.3.14⟩ M8

3.3.11
3.3.12
3.3.13
3.3.14

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Table 3.5.1-1 (Cont'd)

LA1 Bases

A3

A7

EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM (Cont'd)

	Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
Table 3.3.15-1 #12 & <LATER>(3.3C)	b. Low Level SG A or B	4/SG	2/SG	2/SG	1	Note 1
Table 3.3.15-1 #13 & <LATER>(3.3C)	c. Low Pressure, SG A or B	4/SG	2/SG	2/SG	1	Note 1, 19
	d. Loss of Both MFW Pumps and PWR > 10%	4	2	2	1	Note 1
	e. Loss of 4 RC Pumps	4	2	2	1	Note 1, 15
	f. ESAS Actuation Logic Tripped	2	1	2	1	Note 1
	2. SG-A Main Steam Line Isolation					
<LATER> (3.3C)	a. Manual	2	1	2	1	Note 1
	b. Low SG A Pressure	4	2	2	1	Note 1, 19
	3. SG-B Main Steam Line Isolation					
	a. Manual	2	1	2	1	Note 1
	b. Low SG B Pressure	4	2	2	1	Note 1, 19

M7 & LATER

LATER

3.3.15

TABLE 3.5.1-1 (cont'd)

EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM (cont'd)

FUNCTIONAL UNIT	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
OTHER SAFETY RELATED SYSTEMS					
1. Decay heat removal system isolation valve automatic closure and interlock system (3.4.14 LCO)					
a. ~~Reactor coolant pressure instrument channels~~	~~2~~	~~1~~	~~2~~	~~1~~	Notes 1, 5
b. ~~Core flood isolation valve interlocks~~	~~2~~	~~1~~	~~2~~	~~1~~	Notes 1, 5

LA2 TRM

A1

⟨Add 3.4.14 LCO Note⟩ L8

Amendment No. 91

45c

3.4.14

Table 3.5.1-1 (Cont'd)

OTHER SAFETY RELATED SYSTEMS
(Cont'd)

LA1 Bases

A3

A7

	Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met	
Table 3.3.15-1 #11	2. Pressurizer level channels	2	N/A	2	1	Note 10	A1
Table 3.3.15-1 #17	3. Emergency Feedwater flow channels	2/S.G.	N/A	1 2/SG	0	Note 10	M9
	4. RCS subcooling margin monitors	2	N/A	1	0	Note 10	
	5. Electromatic relief valve flow monitor	2	N/A	1	0	Note 11	L15
	6. Electromatic relief block valve position indicator	1	N/A	1	0	Note 12	
	7. Pressurizer code safety valve flow monitors	2/valve	N/A	1/valve	0	Note 10	A10
3.3.8 LCO	8. Degraded Voltage Monitoring						A3
	a. 4.16 KV Emergency Bus Undervoltage	2/Bus	1/Bus	2/Bus	0	Note 14	A7 A1
	b. 460 V Emergency Bus Undervoltage	*1/Bus	1/Bus	1/Bus	0	Notes 13, 14	LA1 Bases
	9. Deleted						A1
Table 3.3.15-1 #9	10. Containment Reactor Building (A1) High Range Radiation Monitoring	2	N/A	2	0	Note 20	LA1 Bases A3
Table 3.3.15-1 #7	11. Containment Pressure - High Range	2	N/A	2	0	Note 21	A1
Table 3.3.15-1 #6	12. Containment Water Level - Wide Range	2	N/A	2	0	Note 21	A7

~~*Two undervoltage relays per bus are used with a coincident trip logic (2-out-of-2)~~ LA1 Bases

3.3.8
3.3.15

Table 3.5.1-1 (cont'd)

OTHER SAFETY RELATED SYSTEMS
(Cont'd)

LA1 Bases

A3

A7

Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
Table 3.3.15-1 PAM #16 13. In core Thermocouples (core-exit thermocouples)	8/core quadrant	N/A	2/core quadrant	0	Note 22 A1
3.3.16 LCO 14. Control Room Radiation Monitors	2	1	2	1	Note 17, 18 A3
Table 3.3.15-1 PAM #5 15. Reactor Vessel Level Monitoring System	2	N/A	2	0	Note 28, 29
Table 3.3.15-1 PAM #3 16. Hot Leg Level Measurement System (HLLMS)	2	N/A	2	0	Note 28, 29 A1
17. Main Steam Line Radiation Monitors	1/steam line	N/A	1/steam line	0	Note 30 LA2 ODCM SAR

LA1 Bases A3 A7

3.3.15
3.3.16

A3

MODE 3 with the CRD breakers open within 6 hrs.

M3 & LATER

TABLE 3.5.1-1 (Cont'd)

3.3.1 Cond. D
3.3.2 Cond. B
& <LATER> (3.3B/C/D & 3.4B)

Notes: 1. Initiate a shutdown using normal operating instructions and place the reactor in the hot shutdown condition within 12 hours if the requirements of Columns 3 and 4 are not met.

3.3.9 Appl.
3.3.10 Appl.

2. When 2 of 4 power range instrument channels are greater than 10% 5% rated power, hot shutdown is not required. — L10

3.3.9 Cond. B

3. When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, ~~hot shutdown is not required.~~ initiate action to restore source range channel within 1 hr. — M7

4. For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours, after which Note 1 applies. — L5

<LATER> (3.3B, 3.4B)

5. If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours. — LATER

3.3.1 RA B.1 & <LATER> (3.3B & 3.3C)

6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel. Otherwise, the actions required by Column 5 shall apply. — A1 & LATER

7. These channels initiate control rod withdrawal inhibits not reactor trips at -10% rated power. Above 10% rated power, those inhibits are bypassed. — LA1 BASES

<LATER> (3.3B)

8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies. — LATER

<LATER> (3.3D)

9. The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies.

10. With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours.

11. With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 3 applies. — LATER

3.3.1
3.3.2
3.3.9
3.3.10

<Add 3.3.6 RA A.1> L4

<Add 3.3.7 RA A.1> L3

A3

TABLE 3.5.1-1 (Cont'd)

3.3.5 RA B.1
3.3.6 RA B.1
& <LATER> (3.3A/C/D & 3.4B)

Notes: 1. ~~Initiate a shutdown using normal operating instructions and~~ place the reactor in ~~the hot shutdown condition~~ MODE 3 within ~~12~~ 6 hours if the requirements of Columns 3 and 4 are not met. M5 & LATER

<LATER> (3.3A)

2. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required. LATER

3. When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, hot shutdown is not required.

4. For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours, after which Note 1 applies.

3.3.5 RA B.2.2
3.3.6 RA B.2
& LATER (3.4B)

5. If the requirements of Columns 3 or 4 cannot be met ~~within an additional 48 hours,~~ place the reactor in the ~~cold shutdown condition~~ MODE 5 within ~~24~~ 36 hours. M4 & LATER

3.3.5 RA A.1
& <LATER> (3.3A, 3.3C)

6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel within one hour. Otherwise, the actions required by Column 5 shall apply. M3 & LATER

<LATER> (3.3A)

7. These channels initiate control rod withdrawal inhibits not reactor trips at ~10% rated power. Above 10% rated power, those inhibits are bypassed. LATER

3.3.7 RA A.2

8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. ~~Hence~~ Declare the associated safety features ~~are~~ inoperable within 1 hour and Specification 3.3 applies. L2 M2 A1

<LATER> (3.3D)

9. The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies. LATER

10. With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours.

11. With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 9 applies.

45e

3.3.5
3.3.6
3.3.7

(A3)

TABLE 3.5.1-1 (Cont'd)

<LATER> (3.3A/B/C, 3.4B) Notes: 1. ~~Initiate a shutdown using normal operating instructions and place the reactor in the hot shutdown condition within 12 hours if the requirements of Columns 3 and 4 are not met.~~ LATER

2. ~~When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.~~

<LATER> (3.3A) 3. ~~When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, hot shutdown is not required.~~ LATER

4. ~~For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours, after which Note 1 applies.~~

<LATER> (3.3B & 3.4B) 5. ~~If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours.~~ LATER

<LATER> (3.3A, B, C) 6. ~~The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel. Otherwise, the actions required by Column 5 shall apply.~~ + LATER

<LATER> (3.3A) 7. ~~These channels initiate control rod withdrawal inhibits not reactor trips at ~10% rated power. Above 10% rated power, those inhibits are bypassed.~~ LATER

<LATER> (3.3B) 8. ~~If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies.~~ LATER

9. ~~The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies.~~ L15

3.3.15 PAMS 11 & 17 RA A.1 10. With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours. L5 L15 M10

11. ~~With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 9 applies.~~ L15

< Add 3.3.15 RA B.1, RA C.1, RA F.1 & F.2 for PAMs #11 & 17 > L5 M10

45e

3.3.15

TABLE 3.5.1-1 (Cont'd)

3.4.14 Cond. B&C & <LATER> (3.3A, 3.3B, 3.3C, 3.3D)

Notes:

1. Initiate a shutdown using normal operating instructions and place the reactor in the hot shutdown condition within 12 hours if the requirements of Columns 3 and 4 are not met. {MODE 3} L8 & LATER M14 A1

<LATER> (3.3A)

2. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

3. When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, hot shutdown is not required.

4. For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours, after which Note 1 applies. LATER

3.4.14 Cond B&C & <LATER> (3.3 B)

5. If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours. {MODE 5} L8 M14 & LATER A1

<LATER> (3.3A, 3.3B & C)

6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel. Otherwise, the actions required by Column 5 shall apply. LATER

<LATER> (3.3A)

7. These channels initiate control rod withdrawal inhibits not reactor trips at -10% rated power. Above 10% rated power, those inhibits are bypassed. LATER

<LATER> (3.3 B)

8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies. LATER

<LATER> (3.3 D)

9. The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies.

10. With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours.

11. With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 9 applies. LATER

3.4.14

<Add 3.3.11 RA E.1> L2 A3

M1

<Add 3.3.11 RA A.1 & B.1>

A1

TABLE 3.5.1-1 (Cont'd)

Notes:

3.3.11 RAs D.1 & F.1 & <LATER> (3.3A/B/D & 3.4B)

1. ~~Initiate a shutdown using normal operating instructions and~~ place the reactor in ~~the hot shutdown~~ MODE 3 ~~condition~~ within ~~12~~ 6 hours if the requirements of Columns 3 and 4 are not met. & LATER M5

2. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required. LATER

<LATER> (3.3A) 3. When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, hot shutdown is not required.

4. For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours, after which Note 1 applies. LATER

<LATER> (3.3B & 3.4B) 5. If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours. & LATER

3.3.11 RA B.2 & <LATER> (3.3A & 3.3B) 6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel within 1 hour. Otherwise, the actions required by Column 5 shall apply. L3

<LATER> (3.3A) 7. These channels initiate control rod withdrawal inhibits not reactor trips at ~10% rated power. Above 10% rated power, those inhibits are bypassed. LATER

<LATER> (3.3B) 8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies. LATER

9. The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies. LATER

<LATER> (3.3D) 10. With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours.

11. With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 9 applies.

3.3.11

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A3

A1

TABLE 3.5.1-1 (Cont'd)

MODE 3 & LATER

3.3.12 RA, C.1 & D.1 (LATER) (3.3A/B/D & 3.4B)

M5

Notes: 1. ~~Initiate a shutdown using normal operating instructions and~~ place the reactor in ~~the hot shutdown condition~~ within ~~12~~ 6 hours ~~if the requirements of Columns 3 and 4 are not met.~~

Addressed on Page 45e-1

2. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

3. When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, hot shutdown is not required.

4. For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours, after which Note 1 applies.

5. If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours.

6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel. Otherwise, the actions required by Column 5 shall apply.

7. These channels initiate control rod withdrawal inhibits not reactor trips at -10% rated power. Above 10% rated power, those inhibits are bypassed.

8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies.

9. The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies.

10. With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours.

11. With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 9 applies.

3.3.12

A3

A1

MODE 3

TABLE 3.5.1-1 (Cont'd)

3.3.13 RA,B.1 & C.1 & (LATER) (3.3A/B/D & 3.4B)

Notes: 1. ~~Initiate a shutdown using normal operating instructions and~~ place the reactor in ~~the hot shutdown~~ & LATER condition within ~~12~~ 6 hours ~~if the requirements of Columns 3 and 4 are not met.~~

M5

Addressed on Page 45e-1

2. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

3. When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, hot shutdown is not required.

4. For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours, after which Note 1 applies.

5. If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours.

6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel. Otherwise, the actions required by Column 5 shall apply.

7. These channels initiate control rod withdrawal inhibits not reactor trips at -10% rated power. Above 10% rated power, those inhibits are bypassed.

8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies.

9. The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies.

10. With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours.

11. With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 9 applies.

3.3.13

A3

TABLE 3.5.1-1 (Cont'd)

<LATER> (3.3D)

12. With the number of operable channels less than required, either return the indicator to operable status within 24 hours, or verify the block valve closed and power removed within an additional 24 hours. If the block valve cannot be verified closed within the additional 24 hours, de-energize the electromatic relief valve power supply within the following 12 hours.

13. Channels may be bypassed for not greater than 30 seconds during reactor coolant pump starts. If the automatic bypass circuit or its alarm circuit is inoperable, the undervoltage protection shall be restored within 1 hour, otherwise, Note 14 applies.

LATER

<LATER> (3.3D + 3.8)

14. With the number of channels less than required, restore the inoperable channels to operable status within 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LATER

Table 3.3.1-1 Function 10 App I Function 9 App I + <LATER> (3.3C)

15. This trip function may be bypassed at up to 10% reactor power.

16. This trip function may be bypassed at up to 45% reactor power.

L3 + LATER

<LATER> (3.3D)

17. With no channel operable, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

18. With one channel inoperable, restore the inoperable channel to operable status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

LATER

<LATER> (3.3C)

19. This function may be bypassed below 750 psig OTSG pressure. Bypass is automatically removed when pressure exceeds 750 psig.

LATER

<LATER> (3.3D)

20. With one channel inoperable, (1) either restore the inoperable channel to operable status within 7 days, or (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.12.5 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

21. With one channel inoperable, restore the inoperable channel to operable status within 30 days or be in hot shutdown within 72 hours unless containment entry is required. If containment entry is required, the inoperable channel must be restored by the next refueling outage. If both channels are inoperable, restore the inoperable channels within 30 days or be in HOT SHUTDOWN within 12 hours.

LATER

A3

TABLE 3.5.1-1 (Cont'd)

<LATER> (3.3D) 12. With the number of operable channels less than required, either return the indicator to operable status within 24 hours, or verify the block valve closed and power removed within an additional 24 hours. If the block valve cannot be verified closed within the additional 24 hours, de-energize the electromatic relief valve power supply within the following 12 hours. — LATER

<LATER> (3.3D) 13. Channels may be bypassed for not greater than 30 seconds during reactor coolant pump starts. If the automatic bypass circuit or its alarm circuit is inoperable, the undervoltage protection shall be restored within 1 hour, otherwise, Note 14 applies. — LATER

<LATER> (3.3D, 3.8) 14. With the number of channels less than required, restore the inoperable channels to operable status within 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. — LATER

+<LATER> (3.3A) 15. This trip function may be bypassed at up to 10% reactor power. — LA1 Bases +LATER

<LATER> (3.3A) 16. This trip function may be bypassed at up to 45% reactor power. — LATER

<LATER> (3.3D) 17. With no channel operable, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

18. With one channel inoperable, restore the inoperable channel to operable status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. — LATER

19. This function may be bypassed below 750 psig OTSG pressure. Bypass is automatically removed when pressure exceeds 750 psig. — LA1 Bases

<LATER> (3.3D) 20. With one channel inoperable, (1) either restore the inoperable channel to operable status within 7 days, or (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.12.5 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

21. With one channel inoperable, restore the inoperable channel to operable status within 30 days or be in hot shutdown within 72 hours unless containment entry is required. If containment entry is required, the inoperable channel must be restored by the next refueling outage. If both channels are inoperable, restore the inoperable channels within 30 days or be in HOT SHUTDOWN within 12 hours. — LATER

3.3-11

A3

TABLE 3.5.1-1 (Cont'd)

12. ~~With the number of operable channels less than required, either return the indicator to operable status within 24 hours, or verify the block valve closed and power removed within an additional 24 hours. If the block valve cannot be verified closed within the additional 24 hours, de-energize the electromatic relief valve power supply within the following 12 hours.~~ L15

3.3.8 RA A.1 — 13. ~~Channels may be bypassed for not greater than 30 seconds during reactor coolant pump starts.~~ LA1 Bases. If the ~~automatic bypass circuit or its alarm circuit~~ is inoperable, the undervoltage protection shall be restored within 1 hour, otherwise, Note 14 applies.

3.3.8 RA A.1 + <LATER> (3.8) — 14. With the number of channels less than required, restore the inoperable channels to operable status ~~within 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~ L1 + LATER, M1

<LATER> (3.3A+3.3C) — 15. ~~This trip function may be bypassed at up to 10% reactor power.~~ LATER

<LATER> (3.3A) — 16. ~~This trip function may be bypassed at up to 45% reactor power.~~ LATER

3.3.16 RA B.1 — 17. With no channel operable, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

3.3.16 RA A.1 — 18. With one channel inoperable, restore the inoperable channel to operable status within 7 days or ~~within the next 6 hours~~ initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. M5

<LATER> (3.3c) — 19. ~~This function may be bypassed below 750 psig OTSG pressure. Bypass is automatically removed when pressure exceeds 750 psig.~~ LATER

3.3.15 PAM #9 RA A.1/B.1 RA C.1/G.1 — 20. With one channel inoperable, (1) either restore the inoperable channel to operable status within ~~7~~ 30 days, or (2) prepare and ~~submit~~ a Special Report ~~to the Commission pursuant to Specification 6.12.5 within 30 days following the event~~ outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within ~~72 hours~~ 7 days in addition to the actions described above. L6, LA1 Bases, L6

3.3.15 PAM #6 & #7 RA A.1 RA C.1 RA F.1 — 21. With one channel inoperable, restore the inoperable channel to operable status within 30 days ~~or be in hot shutdown within 72 hours unless containment entry is required. If containment entry is required, the inoperable channel must be restored by the next refueling outage.~~ If both channels are inoperable, restore the inoperable channels within 30 days ~~or be in HOT SHUTDOWN within 12 hours~~ M10, L7

<Add 3.3.15 RA B.1 + F.2 for PAMs #6 + #7>

3.3.8 3.3.15 3.3.16

{NOTE 14 referenced from CTS 3.7.2.H(2), PAGE 57}

TABLE 3.5.1-1 (Cont'd)

<LATER> (3.3 D) 12. With the number of operable channels less than required, either return the indicator to operable status within 24 hours, or verify the block valve closed and power removed within an additional 24 hours. If the block valve cannot be verified closed within the additional 24 hours, de-energize the electromatic relief valve power supply within the following 12 hours. — LATER

<LATER> (3.3 D) 13. Channels may be bypassed for not greater than 30 seconds during reactor coolant pump starts. If the automatic bypass circuit or its alarm circuit is inoperable, the undervoltage protection shall be restored within 1 hour, otherwise, Note 14 applies. — LATER

3.8.1 RA A.3, F.1, F.2 <LATER> (3.3 D) 14. With the number of channels less than required, restore the inoperable channels to operable status within 72 hours or be in HOT SHUTDOWN within the next ~~8~~ 12 hours and in COLD SHUTDOWN within the following 30 hours. — & LATER — L4

<LATER> (3.3.A & 3.3 C) 15. This trip function may be bypassed at up to 10% reactor power. — LATER

<LATER> (3.3 A) 16. This trip function may be bypassed at up to 45% reactor power. — LATER

<LATER> (3.3 D)

17. With no channel operable, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

18. With one channel inoperable, restore the inoperable channel to operable status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation. — LATER

<LATER> (3.3 C) 19. This function may be bypassed below 750 psig OTSG pressure. Bypass is automatically removed when pressure exceeds 750 psig. — LATER

<LATER> (3.3 D)

20. With one channel inoperable, (1) either restore the inoperable channel to operable status within 7 days, or (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.12.5 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

21. With one channel inoperable, restore the inoperable channel to operable status within 30 days or be in hot shutdown within 72 hours unless containment entry is required. If containment entry is required, the inoperable channel must be restored by the next refueling outage. If both channels are inoperable, restore the inoperable channels within 30 days or be in HOT SHUTDOWN within 12 hours. — LATER

3.8.1

A3

Table 3.5.1-1 (cont'd)

Notes:

3.3.15 PAM#16 RA A.1 RA C.1 RA F.1

22. With the number of operable channels less than two (2) per core quadrant restore the inoperable channel to operable status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours. L5 M10

<Add 3.3.15 RA B.1 & RA F.2 to PAM#16> L5

<Add 3.3.8 Appl> A9

<Add 3.3.8 ACTIONS Note> A8

<Add 3.3.8 RA B.1> L1 M1

Amendment No. 116

45f1

3.3.8
3.3.15

⟨Add 3.3.4 RA C.1, C.2, & C.4⟩ A3

M5 L7

Table 3.5.1-1 (cont'd)

3.3.4 RA C.3 — 23. ~~With the number of operable Electronic (SCR) Trip relays one less than the total number of Electronic (SCR) Trip relays in a channel, restore the inoperable Electronic (SCR) Trip relay to operable status in 48 hours or place the SCRs associated with the inoperable Electronic (SCR) Trip relay in trip in the next hour. With two or more Electronic (SCR) Trip relays inoperable, place all Electronic (SCR) Trip relays associated with that channel in trip in the next hour.~~ ~~This requirement does not apply to the Electronic Trip channels associated with Group 8 Regulating Power Supply.~~ LA1 Bases

24. With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: A1

3.3.4 RA B.1 — a. Within 1 hour:

3.3.4 RA B.2 — 1. Place the inoperable channel in the tripped condition, or

2. Remove power supplied to the control rod trip device associated with the inoperable channel.

~~b. One additional channel may be bypassed for up to 4 hours for surveillance testing and the [in] operable [sic] channel above may be bypassed for up to 30 minutes in any 24-hour period when necessary to test the trip breaker associated with the logic of the channel being tested. The inoperable channel above shall not be bypassed to test the logic of a channel of the trip system associated with the inoperable channel.~~ A13

3.3.4 Cond. A 25. With one of the Control Rod Drive Trip Breaker diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status ~~in 48 hours~~ or place the breaker in trip ~~in the next hour~~. M6

Or remove power from the trip breaker L8

26. ~~Interrupts motor power to the Safety Groups of control rods only.~~ LA1 Bases

27. ~~Deleted~~ A1

3.3.4

A3

Table 3.5.1-1 (cont'd)

3.3.15 PAM's #3 & #5 28. With the number of OPERABLE channels one less than the minimum number of channels required to be OPERABLE:

RA A.1 a. ~~If repairs are feasible,~~ restore the inoperable channel to OPERABLE status within ~~7~~ 30 days ~~or be in at least HOT SHUTDOWN within the next 12 hours.~~ L8

RA B.1 b. ~~If repair is not feasible without shutting down,~~ operation may continue and a special report ~~pursuant to specification 6.12.5~~ shall be submitted ~~to the NRC within 30 days following the failure;~~ ~~describing the action taken, the cause of the inoperability, and the plans and schedules for restoring the channel to OPERABLE status during the next scheduled refueling outage.~~ LA1 Bases

3.3.15 PAMs #3 & #5 29. With the number of OPERABLE channels two less than the minimum channels required to be OPERABLE:

RA C.1 a. ~~If repairs are feasible,~~ restore at least one inoperable channel to OPERABLE status within ~~48 hours~~ 7 days ~~or be in at least HOT SHUTDOWN within the next 12 hours.~~ L8

RA G.1 b. ~~If repair is not feasible without shutting down,~~ operation may continue and a special report ~~pursuant to specification 6.12.5~~ shall be submitted ~~to the NRC within 30 days following the failure;~~ ~~describing the action taken, the cause of the inoperability, and the plans and schedules for restoring the channels to OPERABLE status during the next scheduled refueling outage.~~ LA1 Bases

~~30. With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and: 1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or 2) prepare and submit a Special Report to the Commission pursuant to Specification 6.12.5 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.~~ LA2 ODCM SAR L14

3.1.4
3.1.5
3.1.6

<Add 3.1.4 RA A.1.1 second Completion Time> M10

<Add 3.1.4 RA B.1 Completion Time> M22

~~3~~.5.2 Control Rod Group and Power Distribution Limits

3.1.4 Appl
3.1.5 Appl
3.1.6 Appl
&(LATER) (3.2)

Applicability

This specification applies to power distribution and operation of control rods during ~~power operation~~. MODES 1 and 2. M5 & LATER

Objective

~~To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.~~ A1

Specification

3.1.5 LCO — ~~3~~.5.2.~~1~~
3.1.5 RA A.1.1 & A.1.2
3.1.5 RA B.1.1 & B.1.2
&(LATER) (3.2)

The available shutdown margin shall be greater than or equal to that specified in the COLR with the highest worth control rod fully withdrawn. With the shutdown margin less than that required, ~~immediately~~ initiate and continue boration injection until the required shutdown margin ~~is restored~~. within 1 hr — & LATER, L7

~~3~~.5.2.2 Operation ~~with~~ inoperable rods:

3.1.4 R.A. C.2
3.1.5 R.A. B.2

~~1.~~ ~~Operation with more than one inoperable rod, as defined in Specification 4.7.1 and 4.7.2.3, in the safety or regulating rod groups shall not be permitted.~~ Be in MODE 3 in 6 hours. M7

3.1.4 RA A.1.1
3.1.4 RA A.1.2
3.1.4 RA C.1.1
3.1.4 RA C.1.2

~~2.~~ If a control rod in the regulating or safety rod groups is declared inoperable in the withdrawn position as defined in Specification 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of an available shutdown margin greater than or equal to that specified in the COLR. Boration may be initiated ~~either~~ to the worth of the inoperable rod ~~or until the regulating and transient rod groups are withdrawn to the limits of Specification 3.5.2.5.3, whichever occurs first.~~ ~~Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.~~ L7, M20, L4, L8

3.1.4 RA A.1.1, A.1.2, C.1.1, & C.1.2 Completion Time

3. If within one (1) hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that an available shutdown margin greater than or equal to that specified in the COLR exists combining the worth of ~~the~~ ~~inoperable rod~~ any CONTROL ROD not capable of being fully inserted with each of the other rods, the reactor shall be brought to ~~the Hot Standby condition~~ MODE 3 in 6 hours until this margin is established. M21, L5

3.1.4 RA B.1

and boration to restore SDM has not been initiated

~~4.~~ ~~Following the determination of an inoperable rod as defined in Specification 4.7.1, all remaining rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.~~ L4

3.1.4 RA A.2.2.1 — 5. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60% of the ~~thermal power~~ ~~allowable~~ ~~for the reactor coolant pump combination~~ RTP. A1

<Add 3.1.4 RA A.2.2.1 Completion Time> M9

<Add 3.1.4 RA A.2.2.3 with Note> M19

3.2.1
3.2.2

& ⟨LATER⟩ (3.1)

3.5.2 Control Rod Group and Power Distribution Limits

3.2.1 Appl
3.2.2 Appl

& LATER

M3

Applicability

This specification applies to power distribution and operation of control rods during power operation.

MODES 1 & 2

⟨LATER⟩ (3.1)

Objective

~~To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.~~

LATER

Specification

3.2.1 LCO
3.2.1 RA.D.1
& ⟨LATER⟩ (3.1)

3.5.2.1 The available shutdown margin shall be greater than or equal to that specified in the COLR with the highest worth control rod fully withdrawn. With the shutdown margin less than that required, immediately initiate and continue boration injection until the required shutdown margin is restored.

L8

& LATER

~~3.5.2.2 Operation with inoperable rods:~~

~~1. Operation with more than one inoperable rod, as defined in Specification 4.7.1 and 4.7.2.3, in the safety or regulating rod groups shall not be permitted.~~

~~2. If a control rod in the regulating or safety rod groups is declared inoperable in the withdrawn position as defined in Specification 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of an available shutdown margin greater than or equal to that specified in the COLR. Boration may be initiated either to the worth of the inoperable rod or until the regulating and transient rod groups are withdrawn to the limits of Specification 3.5.2.5.3, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.~~

⟨LATER⟩ (3.1)

LATER

~~3. If within one (1) hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that an available shutdown margin greater than or equal to that specified in the COLR exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the Hot Standby condition until this margin is established.~~

~~4. Following the determination of an inoperable rod as defined in Specification 4.7.1, all remaining rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.~~

~~5. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60% of the thermal power allowable for the reactor coolant pump combination.~~

3.1.4
3.1.6
3.1.8

<Add 3.1.4 RA A.2.1 Completion Time> M23

3.1.4 RA A.2.1
3.1.6 RA A.1

6. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2 operation above 60 percent of the thermal power allowable for the reactor coolant pump combination may continue provided the rods in the group are positioned such that the rod that was declared inoperable is contained within allowable group average position limits of Specification 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5.3.

3.1.4 RA A.2.2.2

3.5.2.3 The worth of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5. A9

LATER (3.2)

3.5.2.4 Quadrant Power Tilt:

3.1.8 LCO

LATER (3.2)

1. Except for physics tests if quadrant power tilt exceeds the tilt limit set in the CORE OPERATING LIMITS REPORT, reduce power so as not to exceed the allowable power level for the existing reactor coolant pump combination less at least 2% for each 1% tilt in excess of the tilt limit. LATER

3.1.8 LCO

2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than the tilt limit except for physics tests, or the following adjustments in setpoints and limit shall be made:

LATER (3.2)

a. The Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.

b. The control rod group and APSR withdrawal limits shall be reduced 2% in power for each 1% tilt in excess of the tilt limit. LATER

c. The reactor power imbalance setpoints shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.

3.1.8 LCO

LATER (3.2)

3. If quadrant power tilt is in excess of 25% except for physics tests or diagnostic testing, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power allowable for the reactor coolant pump combination is restricted as stated in 3.5.2.4.1 above. LATER

4. Quadrant power tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15% of rated power.

<Add 3.1.4 RA A.2.2.2 Completion Time> M16

<Add 3.1.6 RA A.1 Completion Time> M24

<Add 3.1.6 RA A.2 + Completion Time> M25

<Add SR 3.2.4.1 Frequency> M16

3.2.4

<Add 3.2.4 RA A.1.2.1 Comp. Time> M15

<Add 3.2.4 RA A.1.1> M14 L10

<Add 3.2.4 RA A.1.2.2, A.1.2.3, & A.1.2.4 CTs> L1

<LATER> (3.1) LATER

~~6. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2 operation above 60 percent of the thermal power allowable for the reactor coolant pump combination may continue provided the rods in the group are positioned such that the rod that was declared inoperable is contained within allowable group average position limits of Specification 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5.3.~~

~~3.5.2.3 The worth of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.~~

A1

3.5.2.4 Quadrant Power Tilt (QPT) LATER

<LATER> (3.1) A1

3.2.4 LCO

3.2.4 RA A.1.2.1

1. ~~Except for physics tests,~~ if quadrant power tilt exceeds the tilt limit set in the CORE OPERATING LIMITS REPORT, reduce power so as not to exceed the allowable power level for the existing reactor coolant pump ~~combination less at least 2%~~ for each 1% tilt in excess of the tilt limit. THERMAL A1

<LATER> (3.1)

2. Within a period of ~~4~~ 2 hours, the quadrant power tilt shall be reduced to less than the tilt limit ~~except for physics tests~~, or the following adjustments in setpoints and limit shall be made: M13 LATER

3.2.4 RA A.1.2.2 — a. The Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.

3.2.4 RA A.1.2.3 — b. The ~~control~~ regulating rod group ~~and APSR~~ withdrawal ~~limits~~ setpoints given in the COLR shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.

3.2.4 RA A.1.2.4 — c. The ~~reactor~~ Operational power imbalance setpoints given in the COLR shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.

A4 LA1 COLR

3.2.4 Cond. D

<LATER> (3.1)

3. If quadrant power tilt is in excess of ~~25%~~ ~~except for physics tests or diagnostic testing~~, the reactor will be placed ~~in the hot shutdown condition~~. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power ~~allowable~~ ~~for the reactor coolant pump combination~~ is restricted as stated in 3.5.2.4.1 above.

≤ 20% RTP within 4 hours.

L3 LATER A1 M6

SR 3.2.4.1

3.2.4 Appl

4. Quadrant power tilt shall be monitored on a minimum frequency of once every ~~two hours~~ 7 days during ~~power operation~~ MODE 1 above ~~15% of rated power~~ 20% RTP.

L12 A9

the maximum limit specified in the COLR

<Add 3.2.4 RA A.2> M1

<Add 3.2.4 Condition B> M1

<Add 3.2.4 Condition C> M1

3.1.5
3.1.8
3.1.9

3.5.2.5 Control rod positions:

3.1.5 LCO NOTE

1. Technical Specification 3.1.3.5 (safety rod withdrawal) does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2. A5

<LATER> (3.2)

2. ~~Operating rod group overlap shall be 20% ±5 between two sequential groups,~~ except for physics tests. LATER

3.1.8 LCO
3.1.9 LCO

3.5.2.5 Control rod positions:

<LATER> (3.1)

~~1. Technical Specification 3.1.3.5 (safety rod withdrawal) does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.~~ LATER

3.2.1 LCO 2. Operating rod group overlap shall be ~~20% ±5~~ between two sequential groups, ~~except for physics tests.~~ LA1 COLR LATER

within the limits specified in the COLR

<LATER> (3.1)

<Add SR 3.2.1.1> M5

<Add SR 3.2.1.2> M5

<Add 3.2.1 RA A.1 with Note> M10

3.1.8
3.1.9

3.1.8 LCO
3.1.9 LCO

⟨LATER⟩ (3.2)

3. Except for physics tests or exercising control rods, the control rod position setpoints are specified in the CORE OPERATING LIMITS REPORT for 4, 3, AND 2 pump operation. If the applicable control rod position setpoints are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 4 hours.

LATER

3.1.8 LCO
3.1.9 LCO

⟨LATER⟩ (3.2)

4. Except for physics tests or exercising axial power shaping rods (APSRs), the limits for APSR position are specified in the CORE OPERATING LIMITS REPORT.

With the APSRs outside the specified limit provided in the CORE OPERATING LIMITS REPORT, corrective measures shall be taken immediately to achieve the correct position. Acceptable APSR positions shall be attained within 4 hours.

LATER

⟨LATER⟩ (3.2)

3.5.2.6 Reactor Power Imbalance:

1. Reactor power imbalance shall be monitored on a frequency not to exceed 2 hours during power operation above 40% rated power.

3.1.8 LCO

2. Except for physics tests, reactor power imbalance shall be maintained within the envelope defined by the CORE OPERATING LIMITS REPORT.

3. If the reactor power imbalance is not within the envelope defined by the CORE OPERATING LIMITS REPORT, corrective measures shall be taken to achieve an acceptable reactor power imbalance.

4. If an acceptable reactor power imbalance is not achieved within 4 hours, reactor power shall be reduced until reactor power imbalance setpoints are met.

LATER

⟨LATER⟩ (3.2)

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the Superintendent.

LATER

Bases

The reactor power imbalance envelope defined in the CORE OPERATING LIMITS REPORT is based on either LOCA analyses (which have defined the maximum linear heat rate (see CORE OPERATING LIMITS REPORT), such that the maximum cladding temperature will not exceed the Final Acceptance Criteria) or loss of forced reactor coolant flow analysis (such that the hot fuel rod does not experience departure from nucleate boiling condition). Corrective measures will be taken immediately should the indicated quadrant power tilt, control rod position, or reactor power imbalance be outside their specified boundaries. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA or loss of forced reactor coolant flow occur is highly improbable because all of the power distribution parameters (quadrant power tilt, rod position, and reactor power imbalance) must be at their limits while

A2

<Add 3.2.2 RA A.1 and Note> M11

<Add 3.2.2 Condition B> M2

<Add SR 3.2.2.1> M5

<Add 3.2.3 RA A.1> M12

3.2.1 3.2.2 3.2.3

3.2.1 LCO Note

<Add 3.2.1 RA E.1> M7

<LATER> (3.1)

3.2.1 LCO

3. ~~Except for physics tests~~ ~~or~~ exercising ~~control rods~~, the CAPS control rod position setpoints are specified in the CORE OPERATING LIMITS REPORT for 4, 3, AND 2 pump operation. M4

3.2.1 RA D.2.1 & D.2.2, RA A.2, RA B.1, RA C.1

If the applicable CAPS control rod position setpoints are exceeded, corrective measures shall be taken immediately to achieve an acceptable CAPS control rod position. Acceptable CAPS control rod positions shall be attained within 4 hours. L6 LATER

<LATER> (3.1)

3.2.2 LCO

4. ~~Except for physics tests~~ ~~or~~ ~~exercising axial power shaping rods (APSRs)~~, the limits for APSR position are specified in the CORE OPERATING LIMITS REPORT. A6

3.2.2 RA A.2

With the APSRs outside the specified limit provided in the CORE OPERATING LIMITS REPORT, corrective measures shall be taken immediately to achieve the correct position. Acceptable APSR positions shall be attained within ~~4~~ 24 hours. L4

3.5.2.6 AXIAL ~~Reactor~~ CAPS Power ~~/~~ Imbalance: Operating Limits A1

SR 3.2.3.1

1. AXIAL ~~Reactor~~ CAPS power ~~/~~ imbalance shall be monitored on a frequency not to exceed ~~2~~ 12 hours during ~~power operation~~ MODE 1 above 40% ~~rated power~~ RTP. L9 A1

3.2.3 APPL.

<LATER> (3.1)

3.2.3 LCO

2. ~~Except for physics tests~~, AXIAL ~~reactor~~ CAPS power ~~/~~ imbalance shall be maintained within the ~~envelope defined by~~ limits specified in the CORE OPERATING ~~LIMITS REPORT~~ COLR. LATER A1

3.2.3 RA A.2

3. If the AXIAL ~~reactor~~ CAPS power ~~/~~ imbalance is not within the envelope defined by the CORE OPERATING LIMITS REPORT, corrective measures shall be taken to achieve an acceptable reactor power imbalance. L5

3.2.3 RA B.1

4. If an 24 acceptable AXIAL ~~reactor~~ CAPS power ~~/~~ imbalance is not achieved within ~~4~~ hours, reactor power shall be reduced until ~~reactor power imbalance setpoints are met~~. THERMAL POWER is ≤ 40% RTP within the following 4 hours. A3 M8

~~3.5.2.7~~ ~~The control rod drive patch panels shall be locked at all times with limited access to be authorized by the Superintendent.~~ LA1 SAR

Bases

The reactor power-imbalance envelope defined in the CORE OPERATING LIMITS REPORT is based on either LOCA analyses (which have defined the maximum linear heat rate (see CORE OPERATING LIMITS REPORT), such that the maximum cladding temperature will not exceed the Final Acceptance Criteria) or loss of forced reactor coolant flow analysis (such that the hot fuel rod does not experience a departure from nucleate boiling condition). Corrective measures will be taken immediately should the indicated quadrant power tilt, control rod position, or reactor power imbalance be outside their specified boundaries. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA or loss of forced reactor coolant flow occur is highly improbable because all of the power distribution parameters (quadrant power tilt, rod position, and reactor power imbalance) must be at their limits while A2

3.1.8
3.1.9

simultaneously all other engineering and uncertainty factors are also at their limits.* Conservatism is introduced by application of:

a. Nuclear uncertainty factors.
b. Thermal calibration.
c. Fuel densification effects.
d. Hot rod manufacturing tolerance factors.
e. Fuel rod bowing.

The 20 ±5 percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower parts of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Regulating
8	APSR (axial power shaping bank)

A2

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full-out position (1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% Δk/k at rated power. These values have been shown to be safe by the safety analysis (2) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% Δk/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% Δk/k at beginning of life, hot zero power, would result in a lower transient peak thermal power and therefore less severe environmental consequences than a 0.65% Δk/k ejected rod worth at rated power.

Control rod Groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 20%. The normal position at power is for Groups 6 and 7 to be partially inserted.

---

*Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument and calibration errors. The method used to define the operating limits is defined in plant operating procedures.

3.1.8
3.1.9

The quadrant power tilt limits set forth in the CORE OPERATING LIMITS REPORT have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6. These limits in conjunction with the control rod position setpoints in the CORE OPERATING LIMITS REPORT, ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

A2

The quadrant power tilt limits and reactor power imbalance setpoints in the CORE OPERATING LIMITS REPORT, apply when using the plant computer to monitor the limits. The 2-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service. Additional uncertainty is applied to the limits when other monitoring methods are used.

During the physics testing program, the high flux trip setpoints are administratively set as follows to ensure that an additional safety margin is provided.

Test Power	Trip Setpoint %
0	<5
15	50
40	50
50	60
75	85
>75	105.5

REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2

3.2.4

The quadrant power tilt limits set forth in the CORE OPERATING LIMITS REPORT have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6. These limits in conjunction with the control rod position setpoints in the CORE OPERATING LIMITS REPORT, ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

The quadrant power tilt limits and reactor power imbalance setpoints in the CORE OPERATING LIMITS REPORT, apply when using the plant computer to monitor the limits. The 2-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service. Additional uncertainty is applied to the limits when other monitoring methods are used.

A2

During the physics testing program, the high flux trip setpoints are administratively set as follows to ensure that an additional safety margin is provided.

Test Power	Trip Setpoint %
0	<5
15	50
40	50
50	60
75	85
>75	105.5

REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2

3.3.5

3.5.3 ~~Safety Features Actuation System Setpoints~~

~~**Applicability**~~

~~This specification applies to the safety features actuation system actuation setpoints.~~

~~**Objective**~~

~~To provide for automatic initiation of the safety features actuation system in the event of a breach of reactor coolant system integrity.~~

~~**Specification**~~

~~The safety features actuation setpoints and permissible bypasses shall be as follows:~~

A1

Allowable Values

A1

LA1 Bases

	Functional Unit	Action	~~Setpoint~~	
Table 3.3.5-1 Parameter 1	~~High~~ Reactor Building Pressure* - High High	Reactor Building Spray	≤30 psig	(44.7 psia)
Parameter 2	Reactor Building Pressure High	High Pressure Injection	≤ 4 psig	(18.7 psia)
		Start of Reactor Building Cooling and Reactor Building Isolation	≤ 4 psig	(18.7 psia)
		Reactor Bldg. Ventilation	≤ 4 psig	(18.7 psia)
		Low Pressure Injection	≤ 4 psig	(18.7 psia)
		Penetration Room Ventilation	≤ 4 psig	(18.7) psia)
Parameter 3	Low Reactor Coolant System Pressure**	High Pressure Injection	≥ 1585 psig	
		Low Pressure Injection	≥ 1585 psig	
		Start of Reactor Building Cooling and Reactor Building Isolation	≥ 1585 psig	

~~*May be bypassed during reactor building leak rate test.~~ A12

~~**May be bypassed below 1750 psig and is automatically reinstated above 1750 psig.~~ L1

~~With the safety features actuation setpoints less conservative than the above values, declare the channel inoperable and apply the applicable Action requirements of Table 3.5.1-1 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the trip setpoint value.~~ A1

A13

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached in adequate time in the event of a DBA, cover a spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

Low Reactor Coolant System Pressure

The basis for the 1585 psig low reactor coolant pressure setpoint for high and low pressure injection initiation is to establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation. (1)

A2

REFERENCES

(1) FSAR, Section 14.2.2.5

(2) B&W Calculation 32-1158581

LAR

A13

3.2.3
3.2.4

3.5.4 ~~Incore Instrumentation~~

~~Applicability~~

~~Applies to the operability of the incore instrumentation system.~~

~~Objective~~

~~To specify the functional and operational requirements of the incore instrumentation system.~~

A1

Specification

~~Above 80 percent of operating power determined by the reactor coolant pump combination (Table 2.3-1)~~ ~~at least 23 individual incore detectors shall be operable to check gross core power distribution and to assist in the periodic calibration of the out-of-core detectors in regard to the core imbalance trip limits. The detectors shall be arranged as follows and may be a part of both basic arrangements.~~

M18

~~3.5.4.1 Axial Imbalance~~

~~A. Three detectors, one in each of three strings shall lie in the same axial plane with one plane in each axial core half.~~

~~B. The axial planes in each core half shall be symmetrical about the core mid-plane.~~

~~C. The detector shall not have radial symmetry.~~

~~3.5.4.2 Radial Tilt~~

~~A. Two sets of four detectors shall lie in each core half. Each set of four shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.~~

~~B. Detectors in the same plane shall have quarter core radial symmetry.~~

LA1
BASES

~~With the incore detector system inoperable, do not use the system for the above applicable monitoring function. The provisions of Specifications 3.0.3 are not applicable.~~

M18

~~Bases~~

~~A system of 52 incore flux detector assemblies with 7 detectors per assembly has been provided primarily for fuel management purposes. The system includes data display and record functions and is also used for out-of-core nuclear instrumentation calibration and for core power distribution verification.~~

~~A. The out-of-core nuclear instrumentation calibration includes:~~

~~1. Calibration of the split detectors at initial reactor startup during the power escalation program, and periodically thereafter.~~

A2

2. A comparison check with the incore instrumentation in the event one of the four out-of-core power range detector assemblies gives abnormal readings during operation.

3. Confirmation that the out-of-core axial power splits are as expected.

B. Core power distribution verification includes:

1. Measurement at low power initial reactor startup to check that power distribution is consistent with calculations.

2. Subsequent checks during operation to insure that power distribution is consistent with calculations.

3. Indication of power distribution in the event that abnormal situations occur during reactor operation.

C. The safety of unit operation at or below 80 percent of operating power(1) for the reactor coolant pump combinations without the core imbalance trip system has been determined by extensive 3-D calculations. This will be verified during the physics startup testing program.

D. The minimum requirement for 23 individual incore detectors is based on the following: A2

1. An adequate axial imbalance indication can be obtained with 9 individual detectors. Figure 3.5.4-1 shows a typical set of three detector strings with 3 detectors per string that will indicate an axial imbalance. The three detector strings are the center one, one from the inner ring of symmetrical strings and one from the outer ring of symmetrical strings.

2. Figure 3.5.4-2 shows a typical detection scheme which will indicate the radial power distribution with 16 individual detectors. The readings from 2 detectors in a radial quadrant at either plane can be compared with readings from the other quadrants to measure a radial flux tilt.

3. Figure 3.5.4-3 combines Figures 3.5.4-1 and 3.5.4-2 to illustrate a typical set of 23 individual detectors that can be specified as a minimum for axial imbalance determination and radial tilt indication, as well as for the determination of gross core power distributions. Startup testing will verify the adequacy of this set of detectors for the above functions.

E. At least 23 specified incore detectors will be operable to check power distribution above 80 percent power determined by reactor coolant pump combination. These incore detectors will be read out either on the computer or on a recorder. If a set of 23

3.2.3
3.2.4

~~detectors in specified locations is not operable, power will be decreased to or below 80 percent for the operating reactor coolant pump combination.~~

~~REFERENCE~~

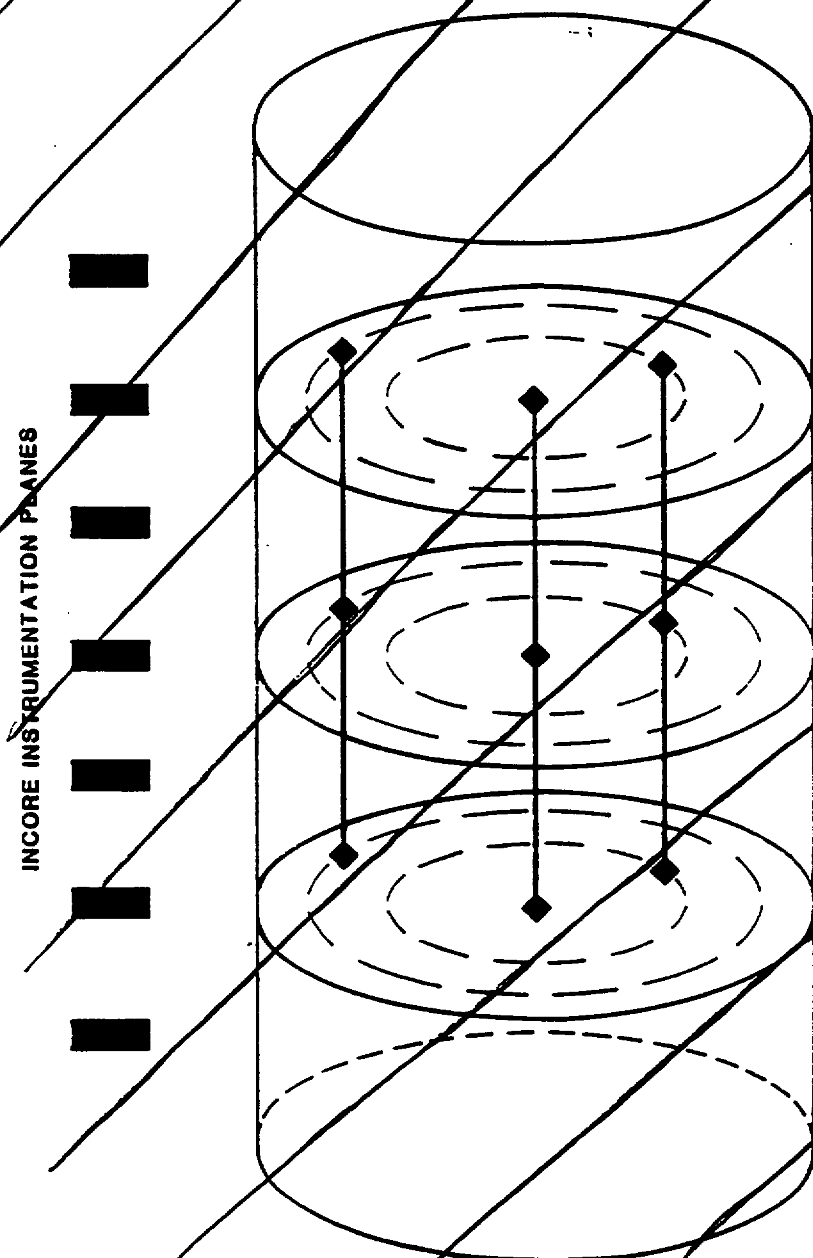
~~(1) FSAR, Section 4.1.1.3~~

A2

3.2.3
3.2.4

~~INCORE INSTRUMENTATION SPECIFICATION~~
~~AXIAL IMBALANCE INDICATION~~

~~Figure No. 3.5.4-1~~

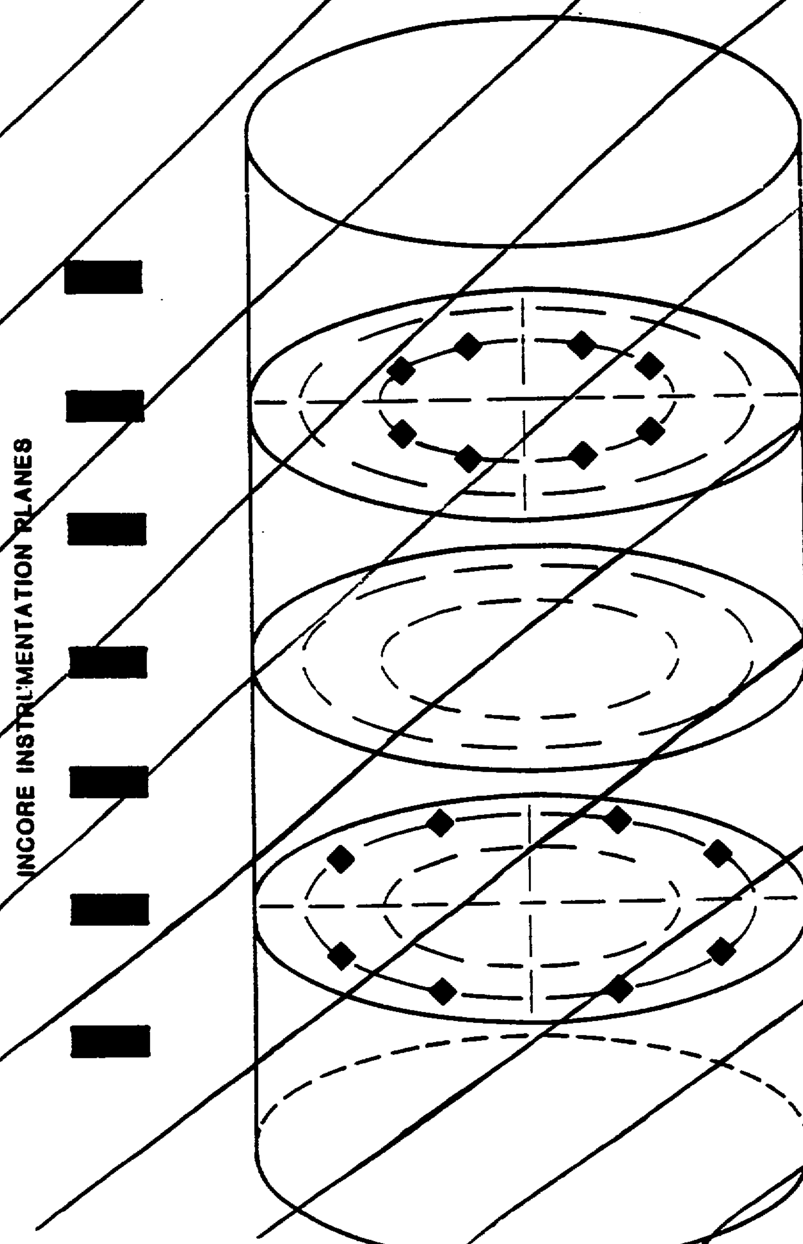


(A2)

3.2.3
3.2.4

INCORE INSTRUMENTATION SPECIFICAITON
RADIAL FLUX TILT INDICATION

Figure No. 3.5.4-2

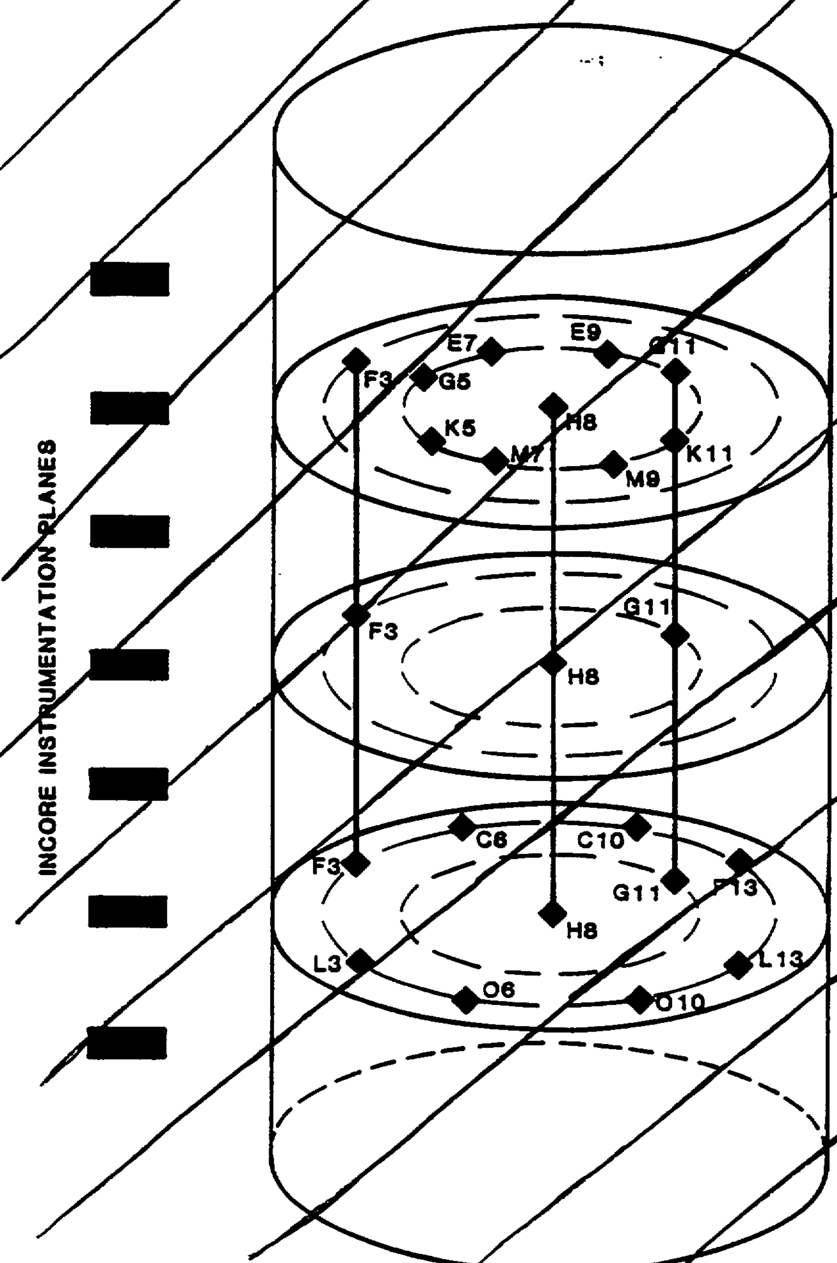


A2

3.2.3
3.2.4

INCORE INSTRUMENTATION SPECIFICATION

Figure No. 3.5.4-3



A2

3.6.1
3.6.2
3.6.3
3.6.4

3.6 REACTOR BUILDING

Applicability

Applies to the operability of the reactor building. (A1)

Objective

To assure reactor building operability.

Specification

3.6.1 LCO
3.6.1 APPL
3.6.2 APPL
3.6.3 APPL

3.6.1 The reactor building shall be operable in MODES 1,2,3,4 whenever all three (3) of the following conditions exist: (M1)

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200°F or greater.

c. Nuclear fuel is in the core.

3.6.2 Act Note 3
3.6.1 RA A.1/B.1/B.2
3.6.2 RA A.1/D.1/D.2
3.6.3 RA D.1/D.2

With the reactor building inoperable, restore the reactor building to operable status within one hour or be in at least Hot Standby MODE 3 within the next 6 hours and in Cold Shutdown MODE 5 within the following 30 hours. (M2) (M22) (A1)

⟨LATER (3.9)⟩

3.6.2 Reactor building integrity shall be maintained when the reactor coolant system is open to the reactor building atmosphere and the requirements for a refueling shutdown are not met. The provisions of Specification 3.0.3 are not applicable. — LATER

3.6.3 Positive reactivity insertions which would result in the reactor being subcritical by less than 1% Δk/k shall not be made by control rod motion or boron dilution whenever reactor building integrity is not in force. The provisions of Specification 3.0.3 are not applicable. (A3)

3.6.4 LCO
3.6.4 APPL
3.6.4 COND A
3.6.4 RA B.1/B.2

3.6.4 MODES 1,2,3,4 The reactor shall not be taken critical or remain critical if the reactor building internal pressure exceeds 3.0 psig or a vacuum of 5.5 inches Hg -1.0 psig. With the reactor critical, restore the containment pressure to within its limits within one hour or be in at least Hot Standby MODE 3 within the next 6 hours and in Cold Shutdown MODE 5 within the following 30 hours. (M13) (M14) (M2) (A1)

3.6.5 Prior to criticality following a refueling shutdown, a check shall be made to confirm that all manual reactor building isolation valves which should be closed are closed and locked as required. The provisions of Specification 3.0.3 are not applicable. (L20)

if not locked, sealed, secured (M16)

SR 3.6.3.2: (for outside RB): Every 31 days

SR 3.6.3.3: (for inside RB): Prior to entering MODE 4 from MODE 5 if not performed in previous 92 days (M4)

⟨CTS INSERT 54A⟩

<CTS INSERT 54A>

<Add 3.6.2 ACTIONS Notes 1, 2, & 3> A8

<Add 3.6.2 Condition A Note 1> A9

<Add 3.6.2 Required Action A.2> M11

<Add 3.6.2 Required Action A.3 & Note> M11

<Add 3.6.2 Condition B with 2 Notes, B.1, B.2 & B.3> M12

<Add 3.6.2 Condition C with RA's C.1, C.2, C.3, & CT's> M12

<Add SR 3.6.3.1 Note & SR 3.6.3.2 Note> L12

<Add SR 3.6.4.1> M13

3.9.3

3.6 REACTOR BUILDING

Applicability

Applies to the operability of the reactor building.

Objective

To assure reactor building operability.

<LATER> (3.6)

Specification

3.6.1 The reactor building shall be operable whenever all three (3) of the following conditions exist: —LATER

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200°F or greater.

c. Nuclear fuel is in the core.

With the reactor building inoperable, restore the reactor building to operable status within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.6.2 Reactor building integrity shall be maintained when the reactor coolant system is open to the reactor building atmosphere and the requirements for a refueling shutdown are not met. The provisions of Specification 3.0.3 are not applicable. L4

3.6.3 Positive reactivity insertions which would result in the reactor being subcritical by less than 1% Δk/k shall not be made by control rod motion or boron dilution whenever reactor building integrity is not in force. The provisions of Specification 3.0.3 are not applicable.

<LATER> (3.6)

3.6.4 The reactor shall not be taken critical or remain critical if the reactor building internal pressure exceeds 3.0 psig or a vacuum of 5.5 inches Hg. With the reactor critical, restore the containment pressure to within its limits within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours. —LATER

3.6.5 Prior to criticality following a refueling shutdown, a check shall be made to confirm that all manual reactor building isolation valves which should be closed are closed and locked, as required. The provisions of Specification 3.0.3 are not applicable.

<ADD: 3.6.3 ACTIONS NOTES 2 & 3> M5
<ADD: 3.6.3 COND A NOTE> A5
<ADD: 3.6.3 RA A.2, & NOTES & CTs> M5
<ADD: 3.6.3 COND B & NOTE>
<ADD: 3.6.3 ACTION C & NOTES>

3.6.3

MODES 1,2,3,4 M3

3.6.3 APPL

3.6.6 If, ~~while the reactor is critical~~, a reactor building isolation valve is determined to be inoperable in a position other than the closed position, the other reactor building isolation valve (except for check valves) in the line shall be tested to insure operability. If the inoperable valve is not restored within 48 hours, the reactor shall be brought to the cold shutdown condition within an additional 24 hours or the operable valve will be closed.

L16 A1

3.6.3 RA A.1
3.6.3 RA D.2
3.6.3 RA D.1

MODE 5 A1

& MODE 3: 6hrs
MODE 5: 36hrs L17

~~Bases~~

~~Included in reactor building operability are both the reactor building integrity as defined in Specification 1.7 and the reactor building structural integrity. Structural integrity limitations as described in the ANO Containment Inspection Program ensure the reactor building will be maintained comparable to the original design standards throughout the facility life span. Visual and other required examinations of tendons, anchorages and surfaces are performed periodically in accordance with station procedures. These procedures embody applicable requirements of the 1992 Edition with the 1992 Addenda of Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code as set forth in 10 CFR 50.55a(g)(6)(ii)(B). Any degradations exceeding the Containment Inspection Program acceptance criteria during inspection surveillances will be reviewed under an engineering evaluation within 60 days of the completion of the inspection to determine what impact the degradation has on overall containment operability, if any.~~

~~The reactor coolant system conditions of cold shutdown assure that no steam will be formed and hence there will be no pressure buildup in the reactor building if the reactor coolant system ruptures.~~

A2

~~The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.~~

~~The reactor building is designed for an internal pressure of 59 psig and an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 110°F and the building is subsequently cooled to an internal temperature of less than 50°F.~~

~~When reactor building integrity is established, the limits of 10 CFR 100 will not be exceeded should the maximum hypothetical accident occur.~~

~~REFERENCE~~

~~FSAR, Section 5.~~

3.8.1
3.8.2
3.8.6

3.7 Auxiliary Electrical Systems

~~Applicability~~

~~Applies to the auxiliary electrical power systems.~~

~~Objectives~~

~~To specify conditions of operation for plant station power necessary to ensure safe reactor operation and combined availability of the engineered safety features.~~ A1

Specifications

MODES 1, 2, 3, & 4

3.8.1 APPL
3.8.2 APPL
3.8.6 APPL

3.7.1 ~~The reactor shall not be heated or maintained above 200°F unless the following conditions are met (except as permitted by Paragraph 3.7.2)~~ A12

3.8.1.a LCO

A. ~~Any one of the following combinations of power sources operable:~~

1. ~~Startup Transformer No. 1 and Startup Transformer No. 2.~~

2. ~~Startup Transformer No. 2 and Unit Auxiliary Transformer provided that the latter one is connected to the 22KV line from the switchyard rather than to the generator bus.~~ LA1 BASES

3.8.6 LCO

B. All 4160 V switchgear, 480 V load centers, 480 V motor control centers and 120 V AC distribution panels in both of the ESAS distribution systems are operable ~~and are being powered from either one of the two startup transformers or the unit auxiliary transformer.~~ L12

3.8.1.b LCO
3.8.2 LCO

C. Both diesel generator sets are operable each with:

SR 3.8.1.4

1. a separate day tank containing a minimum of 160 gallons of fuel,

SR 3.8.2.1

2. a separate emergency storage tank containing a minimum of 138 inches (20,000 gallons) of fuel, L2

SR 3.8.1.5

3. a separate fuel transfer pump, and

3.8.2 LCO

4. a separate starting air ~~compressor~~. subsystem L9

D. DELETED

E. DELETED

3.8.1.a LCO

F. ~~The off-site power undervoltage and protective relaying interlocks associated with required startup transformer power sources shall be operable per Table 3.5.1-1.~~

3.8.1.a LCO

G. ~~The selective load-shed features associated with Startup Transformer No. 2 shall be operable if selected for auto transfer.~~ LA1 BASES

<INSERT CTS 56A>

<CTS INSERT CTS56A>

Add ITS 3.8.2 Actions & Actions Note	Diesel Fuel Oil and Starting Air	L2
Add ITS 3.8.5	Inverters - Operating	M1

3.8.1
3.8.6

<INSERTS CTS 57A & 57B>

3.8.1 APPL
3.8.6 APPL
3.8.6 RA D.1, D.2, E.1
3.8.1 RA F.1, F.2
3.8.1 RA G.1

MODES 1-4

3.7.2

A. The specifications in 3.7.1 may be modified to allow one of the following conditions ~~to exist after the reactor has been heated above 200F~~. Except as indicated in the following conditions, if any of these conditions are not met, ~~a hot shutdown shall be initiated within 12 hours. If the condition is not cleared within 24 hours,~~ the reactor shall be brought to ~~cold shutdown~~ MODE 5 within an additional 24 hours. (L3) (A1) (M3) (A1)

3.8.1 RA A.3 — B. In the event that one of the offsite power sources specified in 3.7.1.A (1 or 2) is inoperable, reactor operation may continue for up to ~~24~~ 72 hours ~~if the availability of the diesel generators is immediately verified~~. (L4) (L3)

3.8.1 RA B.4 — C. Either one of the two diesel generators may be inoperable for up to 7 days ~~in any month~~ provided that during such 7 days the operability of the remaining diesel generator is demonstrated ~~immediately~~ within 24 hours and daily thereafter, ~~there are no inoperable ESF components associated with the operable diesel generator, and provided that the two sources of off-site power specified in 3.7.1.A(1) or 3.7.1.A(2) are available.~~ (LA2) (L3) (L1) (L3)

3.8.1 RA B.3.2
3.8.1 RA B.2

3.8.6 RA A.1, B.1 — D. Any 4160V, 480V, or 120V switchgear, load center, motor control center, or distribution panel in one of the two ESF distribution systems may be inoperable for up to 8 hours, ~~provided that the operability of the diesel generator associated with the operable ESF distribution system is demonstrated immediately and all of the components of the operable distribution system are operable.~~ If the ESF distribution system is not returned to service at the end of the 8 hour period, Specification 3.7.2.A shall apply. (L3)

3.8.6 RA D.1, D.2

E. DELETED

F. DELETED

G. DELETED

H. ~~If the requirements of Specification 3.7.1.G cannot be met, either:~~

~~(1) place all Startup Transformer No. 2 feeder breakers in "pull-to-lock" within 1 hour, restore the inoperable interlocks to operable status within 30 days, or submit within 30 days a Special Report pursuant to Specification 6.12.5 outlining the cause of the failure, proposed corrective action and schedule for implementation; or~~

~~(2) apply the action requirements of Table 3.5.1-1, Note 14.~~ (M8) (L15) (A1)

<see CTS pg 45f>

<CTS INSERT CTS57A>

for ITS 3.8.1 AC Sources - Operating

Add Required Action A.1 M4

Add "10 day" Completion Time for Required Action A.3 M5
and for Required Action B.4

Add Required Action B.3.1 L6

Add Required Action C.2 and Conditions D and E L3

<CTS INSERT CTS57B>

for ITS 3.8.6 Distribution Systems - Operating

Add "16 hour" Completion Time for Required Action A.1 M5
and for Required Action B.1
and for Required Action C.1

3.8.1
3.8.2
3.8.3
3.8.4
3.8.6

<INSERT 3.8.4 ACTIONS NOTE> L5

3.8.3 LCO, 3.8.3 APPL, 3.8.4 APPL — 3.7.3 Both 125 VDC electrical power subsystems shall be operable ~~when the unit is above the cold shutdown condition~~. MODES 1, 2, 3 & 4 A1

A. With one 125 VDC electrical power subsystem inoperable:

<LATER> (5.0) — 1. ~~verify that there are no inoperable safety related components associated with the operable 125 VDC electrical subsystem which are redundant to the inoperable 125 VDC electrical power subsystem,~~ LATER

3.8.1 RA B.2 — 2. verify the operability of the diesel generator associated with the operable 125 VDC electrical subsystem immediately, and

3.8.3 RA A.1 — 3. restore the 125 VDC electrical subsystem to operable status within 8 hours.

3.8.3 RA B.1, B.2 — B. With one 125 VDC electrical power subsystem inoperable, and unable to satisfy the requirements or allowable outage times of 3.7.3.A.1, 3.7.3.A.2, or 3.7.3.A.3, the unit shall be placed in ~~hot shutdown~~ MODE 3 within 12 hours and in ~~cold shutdown~~ MODE 5 within an additional 24 hours. A1

3.8.4 LCO — 3.7.4 Battery cell parameters shall be within limits when the associated DC electrical power subsystems are required to be operable.

3.8.4 COND A — A. With one or more batteries with one or more battery cell parameters not within Table ~~4.6-1~~ 3.8.4-1 Category A or B limits:

3.8.4 RA A.1 — 1. Within 1 hour, verify pilot cell electrolyte level and float voltage meet Table ~~4.6-1~~ Category C limits, A1

3.8.4 RA A.2 — 2. Within 24 hours and once per 7 days thereafter, verify battery cell parameters meet Table ~~4.6-1~~ Category C limits, and

3.8.4 RA A.3 — 3. Within 31 days, restore battery cell parameters to Table ~~4.6-1~~ 3.8.4-1 Category A and B limits.

3.8.4 COND B & RA B.1 — B. With one or more batteries with one or more battery cell parameters not within Table ~~4.6-1~~ Category A or B limits and unable to satisfy the requirements or allowable outage times of 3.7.4.A.1, 3.7.4.A.2, or 3.7.4.A.3, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A. A1

C. With one or more batteries with electrolyte temperature of the pilot cell not within the limits of Specification 4.6.2.8, electrolyte temperature of representative cells not within the limits of Specification 4.6.2.6 or with one or more batteries with one or more battery cell parameters not within Table ~~4.6-1~~ 3.8.4-1 Category C limits, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A.

~~Bases~~

~~The electrical system is designed to be electrically self-sufficient and provide adequate, reliable power sources for all electrical equipment during startup, normal operation, safe shutdown and handling of all emergency situations. To prevent the concurrent loss of all auxiliary power, the various sources of power are independent of and isolated from each other.~~ A2

5.5.15

<LATER> (3.8)

3.7.3 Both 125 VDC electrical power subsystems shall be operable when the unit is above the cold shutdown condition.

A. With one 125 VDC electrical power subsystem inoperable: LATER

5.5.15

1. verify that there are no inoperable safety related components associated with the operable 125 VDC electrical subsystem which are redundant to the inoperable 125 VDC electrical power subsystem, L4

2. verify the operability of the diesel generator associated with the operable 125 VDC electrical subsystem immediately, and

3. restore the 125 VDC electrical subsystem to operable status within 8 hours.

<LATER> (3.8)

B. With one 125 VDC electrical power subsystem inoperable, and unable to satisfy the requirements or allowable outage times of 3.7.3.A.1, 3.7.3.A.2, or 3.7.3.A.3, the unit shall be placed in hot shutdown within 12 hours and in cold shutdown within an additional 24 hours.

3.7.4 Battery cell parameters shall be within limits when the associated DC electrical power subsystems are required to be operable.

A. With one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category A or B limits:

1. Within 1 hour, verify pilot cell electrolyte level and float voltage meet Table 4.6-1 Category C limits,

2. Within 24 hours and once per 7 days thereafter, verify battery cell parameters meet Table 4.6-1 Category C limits, and

3. Within 31 days, restore battery cell parameters to Table 4.6-1 Category A and B limits.

B. With one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category A or B limits and unable to satisfy the requirements or allowable outage times of 3.7.4.A.1, 3.7.4.A.2, or 3.7.4.A.3, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A.

C. With one or more batteries with electrolyte temperature of the pilot cell not within the limits of Specification 4.6.2.8, electrolyte temperature of representative cells not within the limits of Specification 4.6.2.6 or with one or more batteries with one or more battery cell parameters not within Table 4.6-1 Category C limits, declare the associated battery inoperable immediately and perform the required actions of 3.7.3.A.

__Bases__

The electrical system is designed to be electrically self-sufficient and provide adequate, reliable power sources for all electrical equipment during startup, normal operation, safe shutdown and handling of all emergency situations. To prevent the concurrent loss of all auxiliary power, the various sources of power are independent of and isolated from each other. LATER

3.8.1
3.8.2
3.8.3
3.8.4
3.8.6

In the event that the offsite power sources specified in 3.7.1.A (1 or 2) are inoperable, the required capacity of one emergency storage tank plus one day tank (20, 160 gallons) will be sufficient for not less than three and one-half days operation for one diesel generator loaded to full capacity. (ANO-1 FSAR 8.2.2.3) The underground emergency storage tanks are gravity fed from the bulk storage tank and are normally full, while the day tanks are fed from transfer pumps which are capable of being cross connected at their suction and discharges and automatically receive fuel oil when their inventory is less than 180 gallons. Thus, at least a seven day total diesel oil inventory is available onsite for emergency diesel generator operation during complete loss of electric power conditions.

Technical Specification 3.7.2 allows for the temporary modification of the specifications in 3.7.1 provided that backup system(s) are operable with safe reactor operation and combined availability of the engineered safety features ensured.

A2

Technical Specifications 3.7.1.F and 3.7.1.G provide assurance that the Startup Transformer No. 2 loads will not contribute to a sustained degraded grid voltage situation. This will protect ESF equipment from damage caused by sustained undervoltage.

The 125 VDC electrical power system consists of two independent and redundant safety related class 1E DC electrical subsystems. Each subsystem consists of one 100% capacity 125 VDC battery, its associated battery charger, and its distribution network. Additionally, there is one spare battery charger per subsystem, which provides backup service in the event that the preferred battery charger is out of service.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, no operable battery charger, or inoperable battery and no operable associated battery charger), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst-case single failure would, however, result in the complete loss of the remaining 125 VDC electrical power subsystems with attendant loss of ES functions, continued power operation should not exceed 8 hours.

Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational event or a postulated design basis accident. Cell parameter limits are conservatively established, allowing continued DC electrical system function even with Table 4.6-1 Category A and B limits not met.

With one or more cells in one or more batteries not within limits (i.e., Table 4.6-1 Category A limits not met, or Category B limits not met, or Category A and B limits not met) but within the Table 4.6-1 Category C limits, the battery is degraded but has sufficient capacity to perform its intended function. Therefore, the battery is not required to be considered inoperable solely as a result of Category A or B limits not met, and continued operation is permitted for a limited period of time. The pilot cell electrolyte level and float voltage are required to be verified to meet the Table 4.6-1 Category C limits within 1 hour (TS 3.7.4.A.1). These checks will provide a quick representative status of the remainder of the battery cells. Verification that the Table 4.6-1 Category C limits are met (TS 3.7.4.A.2) provides assurance that during the time needed to restore the parameters to within the Category A and B limits, the battery will still be capable of performing its intended function. This verification is repeated at 7 day intervals until the parameters are restored to within Category A and B limits. This periodic verification is consistent with the increased potential to exceed these battery parameter limits during these conditions.

3.8.4

~~With one or more batteries with one or more battery cell parameters outside the Table 4.6-1 Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured. Therefore, the battery must be immediately declared inoperable and the corresponding DC electrical power subsystem must be declared inoperable.~~

~~Additionally, other potentially extreme conditions, such as electrolyte temperature of the pilot cell falling below 60°F, average electrolyte temperature of representative cells falling below 60°F or battery terminal voltage below the limit are also cause for immediately declaring the associated DC electrical power subsystem inoperable.~~

A2

1.1

3.8 FUEL LOADING AND REFUELING

<u>Applicability</u>

LATER

<LATER> (3.9)

Applies to fuel loading and refueling operations.

<u>Objective</u>

To assure that fuel loading, refueling and fuel handling operations are performed in a responsible manner.

<u>Specification</u>

3.8.1 Radiation levels in the reactor building refueling area shall be monitored by instrument RE-8017. Radiation levels in the spent fuel storage area shall be monitored by instrument RE-8009. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

R TRM

<LATER> (3.9)

LATER

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3.a. At least one decay heat removal loop shall be in operation.* Otherwise, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system, and close all containment penetrations providing access from the containment atmosphere to the outside atmosphere within 4 hours.

b. When the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet, two decay heat removal loops shall be operable.**

A8

Otherwise, immediately initiate corrective action to return the required loops to operable status as soon as possible.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required for refueling shutdown.

3.8.5 Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place.

R TRM

<LATER> (3.9)

LATER

*The decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of core alterations.

**The normal or emergency power source may be inoperable for each shutdown cooling loop.

A8

3.9.1
3.9.2
3.9.4
3.9.5

< Add 3.9.5 RA A.2 > — A7

< Add SR 3.9.2.1 > — M6

< Add SR 3.9.2.2 with Note > — M6

< Add 3.9.2 RA B.2 > — M5

< Add 3.9.4 Appl. > — A6

~~3.8 FUEL LOADING AND REFUELING~~

~~Applicability~~

~~Applies to fuel loading and refueling operations.~~

~~Objective~~

~~To assure that fuel loading, refueling and fuel handling operations are performed in a responsible manner.~~

~~Specification~~ — A1

~~3.8.1 Radiation levels in the reactor building refueling area shall be monitored by instrument RE-8017. Radiation levels in the spent fuel storage area shall be monitored by instrument RE-8009. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.~~ — R TRM

3.9.2 LCO b. — 3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, ~~each with continuous indication available~~, whenever core geometry is being changed. ~~When core geometry is not being changed, at least~~ one neutron flux monitor shall be in service. — LA1 TRM — A1

3.9.2 Appl — MODE 6

3.9.2. LCO a.

3.9.4 LCO Note

3.9.4/3.9.5 LCO — 3.8.3.a. ~~At least~~ one decay heat removal loop shall be in operation.* ~~Otherwise~~, suspend all operations involving an increase in the reactor decay heat load ~~or~~ a reduction in boron concentration of the reactor coolant system, ~~and~~ close all containment penetrations providing access from the containment atmosphere to the outside atmosphere within 4 hours. — immediately — A8

3.9.4 R.A. A.2

3.9.4 RA A.1

3.9.4 R.A. A.4

3.9.5 Appl — b. When the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet, two decay heat removal loops shall be operable.** — LATER (3.8)

3.9.5 LCO — M3

3.9.4 RA A.3

3.9.5 RA A.1 — ~~Otherwise~~, immediately initiate corrective action to return the required loops to operable status ~~as soon as possible~~. immediately — A1

MODE 6

3.9.1 Appl — 3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required ~~for refueling shutdown~~ in the COLR. — M4

3.9.1 LCO

~~3.8.5 Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place.~~ — R TRM

3.9.4 LCO NOTE — *The decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period ~~during the performance of core alterations~~. — A4

<LATER (3.8)> — **The normal or emergency power source may be inoperable for each shutdown cooling loop. — LATER

provided no operations are permitted that would cause reduction in RCS boron concentration. — A4

Items on this page also addressed in the following packages: 3.8, 3.9

3R

3.8 FUEL LOADING AND REFUELING

Applicability

Applies to fuel loading and refueling operations.

Objective

To assure that fuel loading, refueling and fuel handling operations are performed in a responsible manner.

Specification

3.8.1 ~~Radiation levels in the reactor building refueling area shall be monitored by instrument RE-8017. Radiation levels in the spent fuel storage area shall be monitored by instrument RE-8009. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.~~ R TRM

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3.a. At least one decay heat removal loop shall be in operation.* Otherwise, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system, and close all containment penetrations providing access from the containment atmosphere to the outside atmosphere within 4 hours.

b. When the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet, two decay heat removal loops shall be operable.**

Otherwise, immediately initiate corrective action to return the required loops to operable status as soon as possible.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required for refueling shutdown.

3.8.5 ~~Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place.~~ R TRM

*The decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of core alterations.

**The normal or emergency power source may be inoperable for each shutdown cooling loop.

3.9.1
3.9.2
3.9.3
3.9.6

<ADD SR 3.9.3.2 with Note> M7
<ADD SR 3.9.6.1> M10
<ADD SR 3.9.3.1> M8

3.9.6 APPL
3.9.3 APPL
3.9.3 LCO b
3.9.3 LCO a
3.9.6 LCO

3.8.6 During the handling of irradiated fuel in the reactor building, at least one door on the personnel and emergency hatches shall be capable* of being closed. The equipment hatch cover shall also be capable* of being closed. At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel. LA1 Bases

3.9.3 LCO c.1
3.9.3 LCO c.2

3.8.7 Isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed. L1

3.8.8 When two irradiated fuel assemblies are being moved simultaneously by the bridges within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times. R TRM

3.9.5 RA A.1
3.9.4 RA A.1, A.2, A.3
3.9.1 Cond A
3.9.2 Cond A
3.9.2 RA B.1
3.9.6 Cond A
3.9.3 Cond A

3.8.9 If any of the above (immediately) specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made. ~~The provisions of Specification 3.0.3 are not applicable.~~ L5 A3 + R TRM A5

3.9.3 LCO c.3
SR 3.9.3.3

3.8.10 The reactor building purge isolation system, including the radiation monitors shall be tested and verified to be operable (every 18 months) ~~within 7 days prior to refueling operations. The provisions of Specification 3.0.3 are not applicable.~~ M9 A5 L3 M13

~~3.8.11 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 100 hours. In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool. The provisions of Specification 3.0.3 are not applicable.~~ LA2 TRM

~~3.8.12 All fuel handling in the Auxiliary Building shall cease upon notification of the issuance of a tornado watch for Pope, Yell, Johnson, or Logan counties in Arkansas. Fuel handling operations in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

~~3.8.13 No loaded spent fuel shipping cask shall be carried above or into the Auxiliary Building equipment shaft unless atmospheric dispersion conditions are equal to or better than those produced by Pasquill Type D stability accompanied by a wind velocity of 2 m/sec. In addition, the railroad spur door of the Turbine Building shall be closed and the fuel handling area ventilation system shall be in operation. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

~~3.8.14 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

~~* Administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors and/or equipment hatch are open, a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of the reactor building, and any obstruction(s) (e.g. cables and hoses) that could prevent closure of an airlock door and the equipment hatch cover be capable of being quickly removed.~~ LA1 Bases (3.9.3)

SR

Items on this page are also addressed in package 39

3.8.6 During the handling of irradiated fuel in the reactor building, at least one door on the personnel and emergency hatches shall be capable* of being closed. The equipment hatch cover shall also be capable* of being closed. At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

3.8.7 Isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed.

3.8.8 When two irradiated fuel assemblies are being moved simultaneously by the bridges within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times. R TRM

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made. The provisions of Specification 3.0.3 are not applicable. R TRM

3.8.10 The reactor building purge isolation system, including the radiation monitors shall be tested and verified to be operable within 7 days prior to refueling operations. The provisions of Specification 3.0.3 are not applicable.

3.8.11 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 100 hours. In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool. The provisions of Specification 3.0.3 are not applicable.

3.8.12 All fuel handling in the Auxiliary Building shall cease upon notification of the issuance of a tornado watch for Pope, Yell, Johnson, or Logan counties in Arkansas. Fuel handling operations in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position. The provisions of Specification 3.0.3 are not applicable. R TRM

3.8.13 No loaded spent fuel shipping cask shall be carried above or into the Auxiliary Building equipment shaft unless atmospheric dispersion conditions are equal to or better than those produced by Pasquill Type D stability accompanied by a wind velocity of 2 m/sec. In addition, the railroad spur door of the Turbine Building shall be closed and the fuel handling area ventilation system shall be in operation. The provisions of Specification 3.0.3 are not applicable. R TRM

3.8.14 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool. The provisions of Specification 3.0.3 are not applicable. R TRM

* Administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors and/or equipment hatch are open, a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of the reactor building, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door and the equipment hatch cover be capable of being quickly removed.

3.7.13
3.7.14

<LATER> (4.0) 3.8.15 Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specification 3.0.3 are not applicable. — LATER

3.7.14 LCO + Appl. SR 3.7.14.1 + <LATER> (4.0) 3.8.16 Storage in Region 2 (as shown on Figure 3.8.1) of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.8.2. In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (non-restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations. The provisions of Specification 3.0.3 are not applicable. LA3 SAR LA3 SAR

3.7.14 RA A.1 Note

3.7.13 LCO + Appl 3.8.17 The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million. L15 A8

3.7.9 LCO + Appl 3.7.10 LCO + Appl 3.8.18 During the handling of irradiated fuel, the control room emergency air conditioning system and the control room emergency ventilation system shall be operable as required by Specification 3.9. A1

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.6 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. A2

The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel at the refueling temperature (normally 140°F), and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.(1)

The requirement to have two decay heat removal loops operable when there is less than 23 feet of water above the core, ensures that a single failure of the operating decay heat removal loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating decay heat removal loop, adequate time is provided to initiate emergency procedures to cool the core.

The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.(2) Although the refueling boron concentration is sufficient to maintain the core keff ≤ 0.99 if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and

<Add 3.7.13 Cond A.> M17

<Add 3.7.14 RA A.1> M17

3.9.1 3.9.4
3.9.2 3.9.5
3.9.3 3.9.6

<LATER> (4.0)

3.8.15 Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specification 3.0.3 are not applicable. LATER

<LATER> (3.7+ 4.0)

3.8.16 Storage in Region 2 (as shown on Figure 3.8.1) of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.8.2. In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (non-restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations. The provisions of Specification 3.0.3 are not applicable. LATER

<LATER> (3.7)

3.8.17 The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million.

3.8.18 During the handling of irradiated fuel, the control room emergency air conditioning system and the control room emergency ventilation system shall be operable as required by Specification 3.9. LATER

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.6 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel at the refueling temperature (normally 140°F), and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.(1) A2

The requirement to have two decay heat removal loops operable when there is less than 23 feet of water above the core, ensures that a single failure of the operating decay heat removal loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating decay heat removal loop, adequate time is provided to initiate emergency procedures to cool the core.

The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.(2) Although the refueling boron concentration is sufficient to maintain the core keff ≤ 0.99 if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and

4.3

4.3.1.1.a 3.8.15 Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. ~~The provisions of Specification 3.0.3 are not applicable.~~ A4 LATER

+<LATER> (3.7)

4.3.1.1.d 4.3.1.1.e 3.8.16 Storage in Region 2 ~~(as shown on Figure 3.8.1)~~ of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.8.2. ~~In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (non-restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations. The provisions of Specification 3.0.3 are not applicable.~~ A4 +LATER

<LATER> (3.7)

<LATER> (3.7) ~~3.8.17 The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million.~~ LATER

<LATER> (3.7) ~~3.8.18 During the handling of irradiated fuel, the control room emergency air conditioning system and the control room emergency ventilation system shall be operable as required by Specification 3.9.~~ LATER

Bases

~~Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.6 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.~~

~~The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel at the refueling temperature (normally 140°F), and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.(1)~~ A2

~~The requirement to have two decay heat removal loops operable when there is less than 23 feet of water above the core, ensures that a single failure of the operating decay heat removal loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating decay heat removal loop, adequate time is provided to initiate emergency procedures to cool the core.~~

~~The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.(2) Although the refueling boron concentration is sufficient to maintain the core keff ≤ 0.99 if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and~~

3.7.13
3.7.14

replacement. The keff with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

A2 + R TRM

The specification requiring testing reactor building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Because of physical dimensions of the fuel bridges, it is physically impossible for fuel assemblies to be within 10 feet of each other while being handled.

<LATER> (3.9)

Per specification 3.8.6, the reactor building personnel and/or emergency airlock doors and the equipment hatch may be open during movement of irradiated fuel in the reactor building provided at least one door of each airlock and the equipment hatch are capable of being closed in the event of a fuel handling accident and the plant is in REFUELING SHUTDOWN with 23 feet of water above the fuel seated within the reactor pressure vessel. Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency airlock doors and the equipment hatch will be closed following evacuation of the reactor building. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface. LATE

Specification 3.8.11 is required as: 1) the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 100 hours [3]; and, 2) to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded during a full core offload.

Specification 3.8.14 will assure that damage to fuel in the spent fuel pool will not be caused by dropping heavy objects onto the fuel. Administrative controls will prohibit the storage of fuel in locations adjoining the walls at the north and south ends of the pool, in the vicinity of cask storage area and fuel tilt pool access gates.

Specifications 3.8.15 and 3.8.16 assure fuel enrichment and fuel burnup limits assumed in the spent fuel safety analyses will not be exceeded.

Specification 3.8.17 assures the boron concentration in the spent fuel pool will remain within the limits of the spent fuel pool accident and criticality analyses.

REFERENCES

(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3

3.9.1
3.9.3
3.9.6

replacement. The keff with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing reactor building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Because of physical dimensions of the fuel bridges, it is physically impossible for fuel assemblies to be within 10 feet of each other while being handled. A2 + R TRM

Per specification 3.8.6, the reactor building personnel and/or emergency airlock doors and the equipment hatch may be open during movement of irradiated fuel in the reactor building provided at least one door of each airlock and the equipment hatch are capable of being closed in the event of a fuel handling accident and the plant is in REFUELING SHUTDOWN with 23 feet of water above the fuel seated within the reactor pressure vessel. Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency airlock doors and the equipment hatch will be closed following evacuation of the reactor building. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface.

Specification 3.8.11 is required as: 1) the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 100 hours (3); and, 2) to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded during a full core offload.

Specification 3.8.14 will assure that damage to fuel in the spent fuel pool will not be caused by dropping heavy objects onto the fuel. Administrative controls will prohibit the storage of fuel in locations adjoining the walls at the north and south ends of the pool, in the vicinity of cask storage area and fuel tilt pool access gates.

Specifications 3.8.15 and 3.8.16 assure fuel enrichment and fuel burnup limits assumed in the spent fuel safety analyses will not be exceeded.

Specification 3.8.17 assures the boron concentration in the spent fuel pool will remain within the limits of the spent fuel pool accident and criticality analyses.

REFERENCES

(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3

Items on this page are also addressed in packages 3.7 and 3.9

SR

replacement. The keff with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing reactor building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Because of physical dimensions of the fuel bridges, it is physically impossible for fuel assemblies to be within 10 feet of each other while being handled.

R
TRM

Per specification 3.8.6, the reactor building personnel and/or emergency airlock doors and the equipment hatch may be open during movement of irradiated fuel in the reactor building provided at least one door of each airlock and the equipment hatch are capable of being closed in the event of a fuel handling accident and the plant is in REFUELING SHUTDOWN with 23 feet of water above the fuel seated within the reactor pressure vessel. Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency airlock doors and the equipment hatch will be closed following evacuation of the reactor building. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface.

Specification 3.8.11 is required as: 1) the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 100 hours (3); and, 2) to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded during a full core offload.

Specification 3.8.14 will assure that damage to fuel in the spent fuel pool will not be caused by dropping heavy objects onto the fuel. Administrative controls will prohibit the storage of fuel in locations adjoining the walls at the north and south ends of the pool, in the vicinity of cask storage area and fuel tilt pool access gates.

R
TRM

Specifications 3.8.15 and 3.8.16 assure fuel enrichment and fuel burnup limits assumed in the spent fuel safety analyses will not be exceeded.

Specification 3.8.17 assures the boron concentration in the spent fuel pool will remain within the limits of the spent fuel pool accident and criticality analyses.

<u>REFERENCES</u>

(1) FSAR, Section 9.5

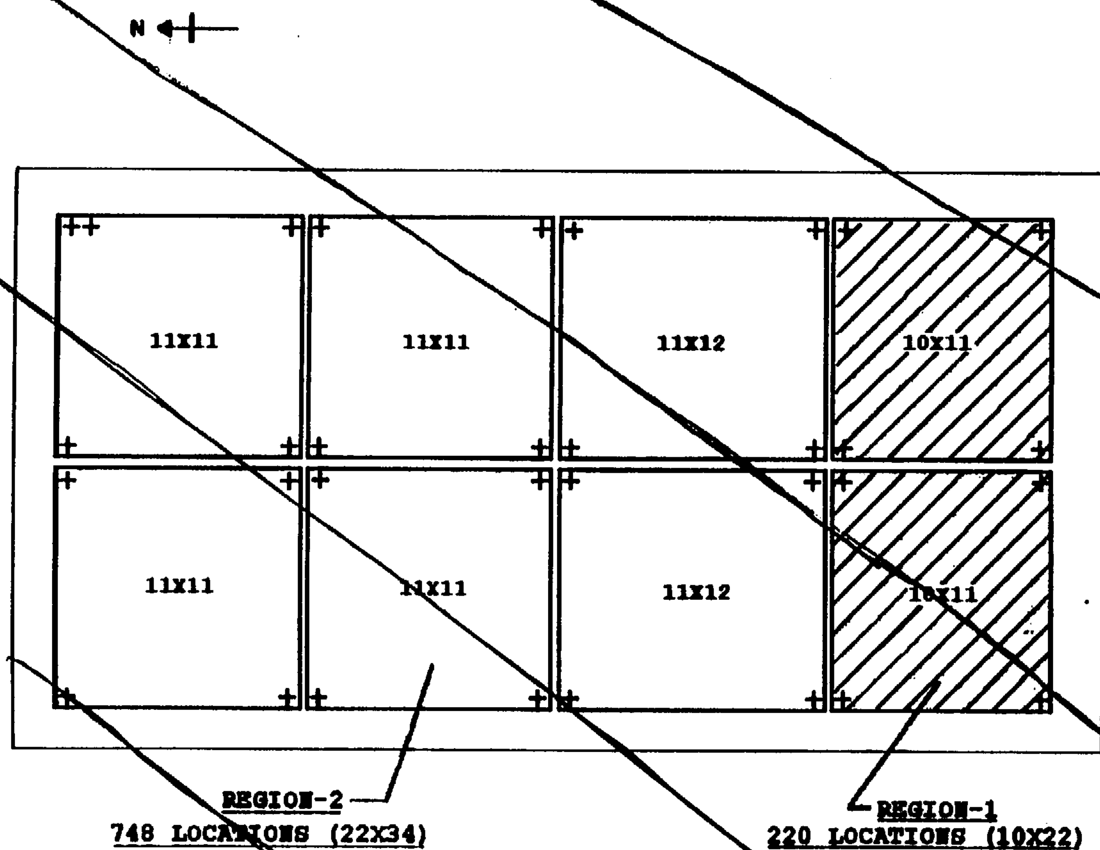
(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3

R
TRM

Spent Fuel Pool Arrangement Unit 1

Figure 3.8.1



LA3
SAR

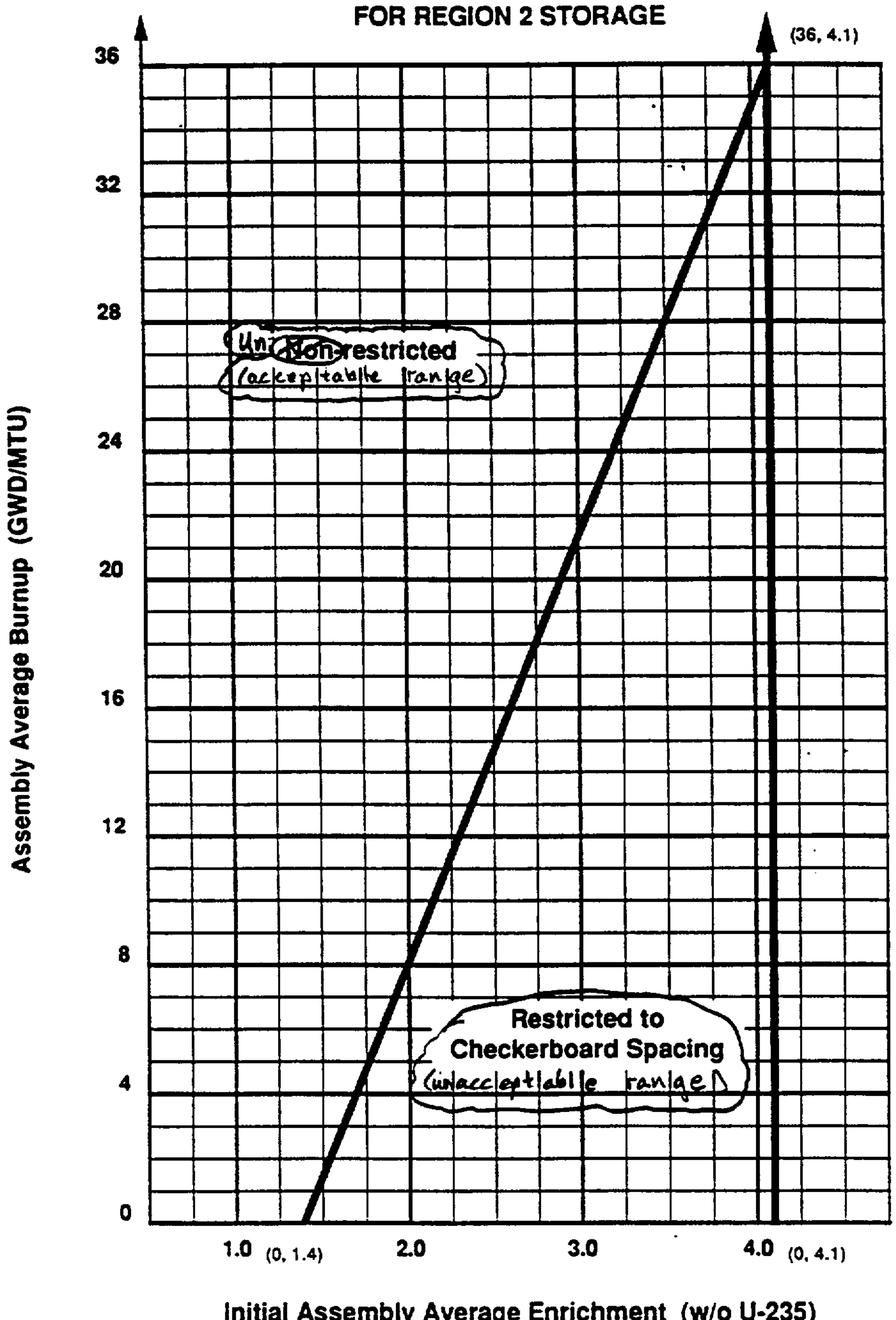
3.7.14

3.7.14

F3.7.14-1

~~FIGURE 3.8.2~~

MINIMUM BURNUP VS. INITIAL ENRICHMENT FOR REGION 2 STORAGE



edit

edit

edit

Figure 3.7.14-1

3.9 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEMS

~~**Applicability**~~

~~Applies to the operability of the control room emergency ventilation and air conditioning systems.~~

~~**Objective**~~

~~To ensure that the control room emergency ventilation and air conditioning systems will perform within acceptable levels of efficiency and reliability.~~ A1

Specification

3.9.1 Control Room Emergency Air Conditioning System

3.7.10 LCO + Appl. — 3.9.1.1 Two independent trains of the control room emergency air conditioning system shall be operable whenever the reactor coolant system is ~~above the cold shutdown condition~~ in MODES 1, 2, 3 or 4 or during handling of irradiated fuel. A1

3.7.10 RA A.1, 3.7.10 RA B.1, 3.7.10 RA B.2 — 3.9.1.2 With one control room emergency air conditioning system inoperable, restore the inoperable system to Operable status within 30 days or be in at least Hot Shutdown within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.9.2 Control Room Emergency Ventilation System

3.7.9 LCO + Appl. — 3.9.2.1 Two independent trains of the control room emergency ventilation system shall be operable whenever the reactor coolant system is ~~above the cold shutdown condition~~ in MODES 1, 2, 3, or 4 or during handling ofirradiated fuel. A1

3.7.9 RA A.1, 3.7.9 RA B.1, B.2 — 3.9.2.2 With one control room emergency ventilation system inoperable, restore the inoperable system to Operable status within 7 days or be in at least Hot Shutdown within the next 6 hours and in Cold Shutdown within the following 30 hours.

<Add 3.7.9 Conds C & D> M18

<Add 3.7.9 Cond E> A6

<Add 3.7.10 Conds C+D> M18

<Add 3.7.10 Cond E> A6

3.7.9
3.7.10

Bases

The control room emergency ventilation and air conditioning system is designed to isolate the combined control rooms to ensure that the control rooms will remain habitable for Operations personnel during and following all credible accident conditions and to ensure that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system. The design configuration of the system is based on limiting the radiation exposure to personnel occupying the control room to 5 REM or less whole body, or its equivalent, in accordance with the requirements of General Design Criteria 19 of Appendix A, 10 CFR 50. A2

Unit 1 and Unit 2 control rooms are a single environment for emergency ventilation and air conditioning concerns. Since the control room emergency ventilation and air conditioning equipment is shared between units, the plant status of both units must be considered when determining applicability of the specification.

Due to the unique situation of the shared emergency ventilation and air conditioning equipment, the components may be cross fed from the opposite unit per predetermined contingency actions/procedures. Unit 1 may take credit for operability of these systems when configured to achieve separation and independence regardless of normal power and/or service water configuration. This will be in accordance with pre-determined contingency actions/procedures.

The control room emergency ventilation system consists of two independent filter and fan trains, two independent actuation channels and the Control Room isolation dampers. The control room dampers isolate the control room within 10 seconds of receipt of a high radiation signal.

If the actuation signal can not start the emergency ventilation recirculation fan, operating the affected fan in the manual recirculation mode and isolating the control room isolation dampers provides the required design function of the control room emergency ventilation system to isolate the combined control rooms to ensure that the control rooms will remain habitable for operations personnel during and following accident conditions. This contingency action should be put in place immediately (within 1 hour) to fully satisfy the design functions of the control room emergency ventilation system.

The control room emergency air conditioning system (CREACS) provides temperature control for the control room following isolation of the control room. It is manually started from the Unit 2 Control Room. The CREACS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment.

With both trains of the control room emergency ventilation and/or emergency air conditioning inoperable, the function of the control room emergency air systems have been lost, requiring immediate action to place the reactor in a condition where the specification does not apply.

3.10 SECONDARY SYSTEM ACTIVITY

~~Applicability~~

~~Applies to the limiting conditions of secondary system activity for operation of the reactor.~~

A1

~~Objective~~

~~To limit the maximum secondary system activity.~~

Specification

3.7.4 LCO
3.7.4 RA A.1
3.7.4 RA A.2

The I-131 dose equivalent of the radioiodine activity in the secondary coolant shall not exceed 0.17 μCi/gm. With the secondary coolant activity in excess of 0.17 μCi/gm I-131, be in at least Hot Standby MODE 3 within 6 hours and in Cold Shutdown MODE 5 within the following 30 hours.

M14

~~Bases~~

A1

A2

~~For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following the rupture of a steam generator tube, a steam line break outside containment and a loss of load incident were considered.~~

~~The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser vacuum pumps. Thus, in the event of a loss of load incident or steam line break, there are only small quantities of these gases which would be released.~~

~~The dose analysis performed to determine the maximum allowable reactor coolant activity assuming the maximum allowable primary to secondary leakage of 1 gpm as given in the Bases for Specification 3.1.4.1 indicated that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam generator tube. For the loss of load incident with a loss of 205,000 pounds of water released to the atmosphere via the relief valves, the resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be 0.6 Rem with the same meteorological and iodine release assumptions used for the steam generator tube rupture as given in the Bases for Specification 3.1.4.1. For the less probable accident of a steam line break, the assumption is made that a loss of 1 x 10⁶ pounds of water or the contents of one loop in the secondary coolant system occurs and is released directly to the atmosphere. Since the water will flash to steam, the total radioiodine activity is assumed to be released to the atmosphere. The resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be less than 28 Rem with the same meteorological assumptions used for the steam generator tube rupture and loss of load incident.~~

3.7.8

3.11 EMERGENCY COOLING POND

Applicability

Applies to the emergency cooling pond.

Objective

To assure the availability of a sufficient supply of cooling water inventory in the emergency cooling pond.

A1

Specification

MODES 1, 2, 3 & 4

M13

3.7.8 LCO
3.7.8 Appl.
SR 3.7.8.1
SR 3.7.8.2
SR 3.7.8.3

3.11.1 The emergency cooling pond shall be operable ~~whenever containment integrity is established as required by Specification 3.6.1 with:~~

1. A minimum contained water volume of 70 acre-feet (equivalent to an indicated water level of 5 feet).

2. An average water temperature of ≤ 100°F.

LA1 Bases

3.7.8 RA A.1
3.7.8 RA A.2

3.11.2 With the requirements of Specification 3.11.1 not satisfied, be in the hot shutdown condition within 6 hours and in the cold shutdown condition within the following 30 hours.

Bases

The requirements of Specification 3.11.1 provide for sufficient water inventory in the emergency cooling pond to mitigate within acceptable limits the effects of a DBA with a concurrent failure of the Dardanelle Reservoir. The minimum water depth takes into account (1) water loss from evaporation due to heat load and climatological conditions, (2) pond bottom irregularities, (3) suction pipe level at the pond and (4) operator action in transferring the service water system from the Dardanelle Reservoi Operator action is credited in the inventory analysis during the transfer of the service water system to the pond. Specifically, pump returns are transferred to the pond shortly after a loss of lake event and pump suctions are transferred later in the event depending on pump bay level. In the time frame between the transfer of the returns and suctions to the pond, lake water is pumped into the pond, increasing level. This additional water is required, along with that maintained by Technical Specifications, to ensure a 64.5 inch pond depth, which corresponds to a 30 day supply of cooling wate

The values are based on worst case initial conditions which could be present considering a simultaneous normal shutdown of Unit 1 and emergency shutdown of Unit 2 following a LOCA in Unit 2, using the ECP as a heat sink. The measured ECP temperature at the discharge from the pond is considered a conservative average of total pond conditions since solar gain, wind speed, and thermal current effects throughout the pond will essentially be at equilibrium conditions under initial stagnant conditions.

A2

3.7

LA3
TRM

3.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability

Applies to byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source and special nuclear radioactive material sources does not exceed allowable limits.

Specification

3.12.1 The source leakage test performed pursuant to Specification 4.14 shall be capable of detecting the presence of 0.005 μCi of radioactive material on the test sample. If the test reveals the presence of 0.005 μCi or more of removable contamination, it shall immediately be withdrawn from use, decontaminated and repaired, or be disposed of in accordance with Commission regulations. Sealed sources are exempt from such leak tests when the source contains 100 μCi or less of beta and/or gamma emitting material or 5 μCi or less of alpha emitting material. The provisions of Specification 3.0.3 are not applicable.

3.12.2 A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.12.5 within 90 days if source leakage tests reveal the presence of ≥0.005 microcuries of removable contamination.

L11

3.12.3 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

A10

3.13 PENETRATION ROOM VENTILATION SYSTEM

Applicability

Applies to the operability of the penetration room ventilation system. A1

Objective

To ensure that the penetration room ventilation system will perform within acceptable levels of efficiency and reliability.

Specification

3.7.11 LCO + Appl SR 3.7.11.12 3.13.1 Two independent circuits of the penetration room ventilation system shall be operable whenever reactor building integrity is required with the following performance capabilities: MODES 1,2,3 and 4 M13 L17

<LATER> (5.0)

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flow (± 10%) on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis from the charcoal adsorber banks shall show the methyl iodide penetration less than 5.0% at velocity within ± 20% of system design, when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%. LATER

c. Fans shall be shown to operate within ± 10% of design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (± 10%).

e. Air distribution shall be uniform within ± 20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system.

SR 3.7.11.3 f. Each circuit of the system shall be capable of automatic initiation.

3.13.2 If one circuit of the penetration room ventilation system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days provided that during such seven days all active components of the other circuit shall be operable. A1

3.7.11 RA A.1

3.13.3 If the requirements of Specifications 3.13.1 and 3.13.2 cannot be met, the reactor shall be placed in the cold shutdown condition within 36 hours. A1

3.7.11 RA B.2

MODE 5

<Add 3.7.11 RA B.1> M20

LAR A7

5.5.11

<ADD PROGRAM DESCRIPTION>

<LATER> (3.7)

~~3.13 PENETRATION ROOM VENTILATION SYSTEM~~

~~Applicability~~

~~Applies to the operability of the penetration room ventilation system.~~

~~Objective~~

~~To ensure that the penetration room ventilation system will perform within acceptable levels of efficiency and reliability.~~

~~Specification~~

~~3.13.1 Two independent circuits of the penetration room ventilation system shall be operable whenever reactor building integrity is required with the following performance capabilities:~~ LATER

5.5.11.a.1 5.5.11.b.1 a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flow (± 10%) on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

5.5.11.c.1 b. The results of laboratory carbon sample analysis from the charcoal adsorber banks shall show the methyl iodide penetration less than 5.0% at velocity within ± 20% of system design, when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

5.5.11.a.1 5.5.11.b.1 5.5.11.d c. Fans shall be shown to operate within ± 10% of design flow.

5.5.11.d d. The pressure drop across ~~the combined HEPA filters~~ other filters in the system and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (± 10%). M5

~~e. Air distribution shall be uniform within ± 20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system.~~ LA2 VFTP

<LATER> (3.7)

~~f. Each circuit of the system shall be capable of automatic initiation.~~

~~3.13.2 If one circuit of the penetration room ventilation system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days provided that during such seven days all active components of the other circuit shall be operable.~~

~~3.13.3 If the requirements of Specifications 3.13.1 and 3.13.2 cannot be met, the reactor shall be placed in the cold shutdown condition within 36 hours.~~

LATER

<ADD: SR 3.0.2 & SR 3.0.3 applicability statement> A3

A10

A2

Bases

The penetration room ventilation system is designed to collect and process potential reactor building penetration leakage to minimize environmental activity levels resulting from post accident reactor building leaks. The system consists of sealed penetration rooms, two redundant filter trains and two redundant fans discharging to the unit vent. The entire system is activated by a reactor building engineered safety features signal and initially requires no operator action. Each filter train is constructed with a prefilter, a HEPA filter and a charcoal adsorber in series. The design flow rate through each of these filters is 2000 scfm, which is significantly higher than the 1.25 scfm maximum leakage rate from the reactor building at a leak rate of 0.1% per day.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of a least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}]}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If one circuit of the penetration room ventilation system is found to be inoperable, there is not an immediate threat to the containment system performance and reactor operation may continue for a limited period of time while repairs are being made.

LAR A7

5.0

A2

Bases

The penetration room ventilation system is designed to collect and process potential reactor building penetration leakage to minimize environmental activity levels resulting from post accident reactor building leaks. The system consists of sealed penetration rooms, two redundant filter trains and two redundant fans discharging to the unit vent. The entire system is activated by a reactor building engineered safety features signal and initially requires no operator action. Each filter train is constructed with a prefilter, a HEPA filter and a charcoal adsorber in series. The design flow rate through each of these filters is 2000 scfm, which is significantly higher than the 1.25 scfm maximum leakage rate from the reactor building at a leak rate of 0.1% per day.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of a least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{(100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis})}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If one circuit of the penetration room ventilation system is found to be inoperable, there is not an immediate threat to the containment system performance and reactor operation may continue for a limited period of time while repairs are being made.

3.14 HYDROGEN RECOMBINERS

Applicability

<LATER> (3.6)

Applies to the operating status of the hydrogen recombiner systems.

LATER

Objective

To ensure that the hydrogen recombiner systems will perform within acceptable levels of efficiency and reliability.

Specification

3.14.1 Two independent hydrogen recombiner systems shall be operable whenever reactor building integrity is required.

3.14.2 Within one hydrogen recombiner system inoperable, restore the inoperable system to operable status within 30 days or the reactor shall be placed in the hot shutdown condition within the next 6 hours.

Table 3.3.15-1, #10 3.14.3 Hydrogen concentration instruments shall be operable.

3.3.15, #10 RA A.1 3.14.4 With one of two hydrogen concentration instruments inoperable restore the inoperable analyzer to OPERABLE status within 30 days ~~or be in at least hot shutdown within the next 6 hours.~~ L9

Bases

The hydrogen recombiner systems are designed to operate as necessary to limit the hydrogen concentration in the reactor building following a Loss of Coolant Accident.

The system is composed of two redundant 100% capacity Internal Electrical Hydrogen Recombiners, manufactured by Westinghouse.

A2

<Add 3.3.15 RA B.1, RA D.1, & RA F.1 for PAM 10> L9

<Add 3.3.15 RA F.2 for PAM 10> M11

<INSERT CTS 66eA>

<Insert CTS 66eA>

Add 3.3.15, ACTIONS Note 1 L4

Add 3.3.15, ACTIONS Note 2 A6

Add 3.3.15, Required Action E.1 for all PAM Functions A1

Add 3.3.15, Condition B Note for all PAM Functions

Add 3.3.15, Condition C Note for all PAM Functions

Add 3.3.15, Condition G Note for PAM Functions 3, 5, & 9

Add 3.3.15, SURVEILLANCES Note for all PAM Functions

Add PAM Functions 1, 2, 8, 12b, 12d, 14 & 20 including all associated LCO, Applicability, ACTIONS, SURVEILLANCES, Notes and Table entries: M7

1. Wide Range Neutron Flux
2. RCS Hot Leg Temperature
8. Automatic Reactor Building Isolation Valve Position
12b. SG "A" Water Level - High Range
12d. SG "B" Water Level - High Range
14. Condensate Storage Tank Level
20. Reactor Building Spray Flow

Add PAM Functions 4, 12a, 12c, & 13 including all associated Applicability, ACTIONS, Notes and Table entries: M7

4. RCS Pressure (Wide Range)
12a. SG "A" Water Level - Low Range
12c. SG "B" Water Level - Low Range
13a. SG "A" Pressure
13b. SG "B" Pressure

Add PAM Functions 18 & 19 including all associated LCO, Applicability, ACTIONS, Notes, Table entries, and SR 3.3.15.1: M7

18. High Pressure Injection Flow
19. Low Pressure Injection Flow

Add 3.3.15, Applicability for PAM Functions 3 & 5 M8

Add 3.3.15, Applicability for PAM Functions 6 & 7

Add 3.3.15, Applicability for PAM Function 10

Add 3.3.15, Applicability for PAM Functions 11 & 17

Add 3.3.15, Applicability for PAM Function 16

Add 3.3.15, Applicability for PAM Function 9 M3

3.6.7

3.14 HYDROGEN RECOMBINERS

~~Applicability~~

~~Applies to the operating status of the hydrogen recombiner systems.~~

~~Objective~~

~~To ensure that the hydrogen recombiner systems will perform within acceptable levels of efficiency and reliability.~~

A1

Specification

LA1 Bases

3.6.7 LCO Appl. 3.14.1 Two independent hydrogen recombiner systems shall be operable whenever reactor building integrity is required. in MODES 1&2 L7

RA.A.1 RA.B.1 3.14.2 With a one hydrogen recombiner system inoperable, restore the inoperable system to operable status within 30 days or the reactor shall be placed in the hot shutdown MODE 3 condition within the next 6 hours. A1 A1

<Later (3.3D)> ~~3.14.3 Hydrogen concentration instruments shall be operable.~~

~~3.14.4 With one of two hydrogen concentration instruments inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least hot shutdown within the next 6 hours.~~ Later

~~Bases~~

~~The hydrogen recombiner systems are designed to operate as necessary to limit the hydrogen concentration in the reactor building following a Loss of Coolant Accident.~~

~~The system is composed of two redundant 100% capacity Internal Electrical Hydrogen Recombiners, manufactured by Westinghouse.~~ A2

<Add 3.6.7 RA A.1 Note> L13

3.7.12

3.15 FUEL HANDLING AREA VENTILATION SYSTEM

Applicability

Applies to the operability of the fuel handling area ventilation system. A1

Objective

To ensure that the fuel handling area ventilation system will perform within acceptable levels of efficiency and reliability.

Specification

3.7.12 LCO +Appl 3.15.1 The fuel handling area ventilation system shall be OPERABLE and in operation whenever irradiated fuel handling operations are in progress in the fuel handling area of the auxiliary building and shall have the following performance capabilities: M3

SR 3.7.12.2

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows (±10%) on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

<LATER> (5.0) b. The results of laboratory carbon sample analysis shall show the methyl iodide penetration less than 5.0% at a velocity within ±20% of system design, when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%. LATE

c. Fans shall be shown to operate within ±10% design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (±10%).

e. Air distribution shall be uniform within ±20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system.

3.7.12 RA A.1 3.15.2 If the requirements of Specification 3.15.1 cannot be met, irradiated fuel movement shall not be started (any irradiated fuel assembly movement in progress may be completed). The provisions of Specification 3.0.3 are not applicable. LA1 Bases

3.7.12 ACTIONS Note

Bases

The fuel handling area ventilation system is designed to filter the auxiliary building atmosphere during fuel handling operations to limit the release of activity should a fuel handling accident occur. The system consists of one circuit containing two exhaust fans and a filter train. The fans are redundant and only one is required to be operating. The filter train consists of a prefilter, a HEPA filter and a charcoal adsorber in series. A2

LAR A7

5.5.11

3.15 FUEL HANDLING AREA VENTILATION SYSTEM

<LATER> (3.7)

Applicability

Applies to the operability of the fuel handling area ventilation system. LATER

Objective

To ensure that the fuel handling area ventilation system will perform within acceptable levels of efficiency and reliability.

Specification

3.15.1 The fuel handling area ventilation system shall be in operation whenever irradiated fuel handling operations are in progress in the fuel handling area of the auxiliary building and shall have the following performance capabilities:

5.5.11.a.1 5.5.11.b.1 a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows (±10%) on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

5.5.11.c.2 b. The results of laboratory carbon sample analysis shall show the methyl iodide penetration less than 5.0% at a velocity within ±20% of system design, when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

5.5.11.a.1 5.5.11.b.1 5.5.11.d c. Fans shall be shown to operate within ±10% design flow.

5.5.11.d d. The pressure drop across ~~the combined HEPA filters~~, other filters in the system, and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (±10%). M5

e. Air distribution shall be uniform within ±20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system. LA3

<LATER> (3.7) 3.15.2 If the requirements of Specification 3.15.1 cannot be met, irradiated fuel movement shall not be started (any irradiated fuel assembly movement in progress may be completed). The provisions of Specification 3.0.3 are not applicable. <LATER>

Bases

The fuel handling area ventilation system is designed to filter the auxiliary building atmosphere during fuel handling operations to limit the release of activity should a fuel handling accident occur. The system consists of one circuit containing two exhaust fans and a filter train. The fans are redundant and only one is required to be operating. The filter train consists of a prefilter, a HEPA filter and a charcoal adsorber in series. A2

3.7.12

A2

~~High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine absorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:~~

~~Allowable Penetration = $\frac{(100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis})}{\text{safety factor of 2}}$~~

~~Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.~~

~~If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.~~

LAR A7

5.0

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine absorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. Acceptable removal efficiency is shown by a methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{(100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis})}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

A2

Items on this page also addressed in the following packages: NA

SR

R

TJRM

3.16 Shock Suppressors (Snubbers)

Applicability

This technical specification applies to all shock suppressors (snubbers). The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Objective

To assure adequate shock suppression protection for primary coolant system piping and any other safety related system or component under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. This is done by assuring the operability of those shock suppressors installed for that purpose.

Specification

3.16.1 With one or more applicable snubbers inoperable, within 72 hours either:

a. Replace or restore the inoperable snubbers to an OPERABLE status and perform an engineering evaluation of the attached components per Specification 4.16.1.f or,

b. Perform a review and evaluation which justifies continued operation with the inoperable snubber(s) and perform an engineering evaluation of the attached component(s) per Specification 4.16.1.f or,

c. Declare the attached system inoperable and follow the appropriate ACTION statement for that system.

Bases

Shock suppressors are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable shock suppressor is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all shock suppressors required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the shock suppressor protection is required only during low probability events, a period of 72 hours is allowed for repairs, replacements or evaluations. If a reveiw and evaluation of an INOPERABLE snubber is preformed and documented to justify continued operation, and provided all design criteria are met with the INOPERABLE snubber, then the INOPERABLE snubber would not need to be restored or replaced. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures.

Items on this page also addressed in the following packages NA

SR

R

7KH1

3.22 REACTOR BUILDING PURGE FILTRATION SYSTEM

Applicability

This specification applies to the operability of the reactor building purge filtration system.

Objective

To assure that the reactor building purge filtration system will perform within acceptable levels of efficiency and reliability.

Specification

3.22.1 The reactor building purge filtration system shall be operable whenever irradiated fuel handling operations are in progress in the reactor building and shall have the following performance capabilities:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows (±10%) on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity within ±20% of system design, 0.05 to 0.15 mg/m^3 inlet methyl iodide concentration, ≥70% R. H. and ≥125F.

c. Fans shall be shown to operate within ±10% design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (±10%).

e. Air distribution shall be uniform within ±20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the reactor building purge filtration system.

3.22.2 If the requirements of Specification 3.22.1 cannot be met, either:

a. Irradiated fuel movement shall not be started (any irradiated fuel assembly movement in progress may be completed); or,

b. Isolate the reactor building purge system.

3.22.3 The provisions of Specification 3.0.3 are not applicable.

Bases

The reactor building purge filtration system is designed to filter the reactor building atmosphere during normal operations for ease of personnel entry into the reactor building. This specification is intended to require the system operable during fuel handling operations, if the system

Items on this page also addressed in the following packages: NA

SR

R

7.2.1.L

is to be used, to limit the release of activity should a fuel handling accident occur. The system consists of one circuit containing a supply and an exhaust fan and a filter train. The filter train consists of a pre-filter, a HEPA filter and a charcoal adsorber in series.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

3.23 REACTOR BUILDING PURGE VALVES

APPLICABILITY

~~This specification applies to the reactor building purge supply and exhaust isolation valves.~~ A1

OBJECTIVE

~~To specify that reactor building isolation purge valves be closed whenever containment integrity is required by TS 3.6.1.~~ LA1 Bases

SPECIFICATION

3.6.3 LCO
Appl.
ACT. Note 1

3.23.1 The reactor building purge ~~supply and exhaust~~ isolation valves shall be closed ~~and handswitch keys removed whenever containment integrity is required by TS 3.6.1.~~ in MODES 1,2,3,4 M1

BASES

~~The reactor building supply and exhaust isolation valves are required to be closed during normal plant operation in order to ensure reactor building integrity. Purging is allowed only when containment integrity is not required by TS 3.6.1.~~ A2

5.5.12

<Add Program description>

5.5.12

3.24 EXPLOSIVE GAS MIXTURE

LA5.

EG&STRMP

Applicability

Applies to the Waste Gas System hydrogen/oxygen analyzers.

Objective

To prevent accumulation of explosive mixture in the waste gas system.

Specification

3.24.1 The Concentration of hydrogen/oxygen shall be limited in the waste gas decay tanks to Region "A" of Figure 3.24-1.

3.24.2 When the hydrogen/oxygen concentration in any of the decay tanks enters Region "B" of Figure 3.24-1, corrective action shall be taken to return the concentration values to Region "A" within 24 hours.

3.24.3 The provisions of Specification 3.0.3 are not applicable.

A2

Bases

These hydrogen/oxygen limits provide reasonable assurance that no hydrogen/oxygen explosion could occur to allow rupture of the waste gas decay tanks. The hydrogen and oxygen limits are based on information in NUREG/CR-2726 "Light Water Reactor Hydrogen Manual".

<Add SR 3.0.2 & SR 3.0.3 applicability statement> A3

5.5.12

<ADD Program Description>

3.25 RADIOACTIVE EFFLUENTS

3.25.1 Radioactive Liquid Holdup Tanks

Applicability: At all times.

Objective: To ensure that the limits of 10 CFR 20 are not exceeded.

LA5
EG+STRMP

Specifications:

3.25.1 A. The quantity of radioactive material contained in each unprotected* outside temporary radioactive liquid storage tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

B. With the quantity of radioactive material exceeding the above limit, immediately suspend all additions of radioactive material to the affected tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluent Release Report pursuant to Specification 6.12.2.6.

C. The provisions of Specification 3.0.3 are not applicable.

Bases:

This specification is provided to ensure that in the event of an uncontrolled release of the contents of the tank* the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in the unrestricted area.

*Tanks included in this specification are those outdoor temporary tanks that 1) are not surrounded by liners, dikes, or walls capable of holding the tank contents, and 2) do not have overflows and surrounding area drains connected to the liquid radwaste treatment system.

A2

<ADD: SR 3.0.2 & SR 3.0.3 applicability statement> A3

5.5.12

<ADD Program Description>

LA5 EG+STRMP

3.25.2 Radioactive Gas Storage Tanks

Applicability: At all times

Objective: To restrict the amount of activity in a radioactive gas holdup tank.

Specifications:

3.25.2 A. The quantity of radioactivity contained in each gas storage tank shall be limited to 300,000 curies noble gases (Xe-133 equivalent).

B. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluent Release Report pursuant to Specification 6.12.2.4.

C. The provisions of Specification 3.0.3 are not applicable.

Bases:

The value of 300,000 curies is a suitable fraction of the quantity of radioactive material which if released over a 2-hour period, would result in a total body exposure to a member of the public at the exclusion area boundary of 500 mrem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

A2

<ADD SR 3.0.2 + SR 3.0.3 applicability statement> A3

ITS

SURVEILLANCE REQUIREMENTS (SR) APPLICABILITY — A1

SR 3.0.1 — 4.0.1 Surveillance Requirements shall be met during the operational modes or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement. — A8

SR 3.0.2 — 4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval. — A9

SR 3.0.1 — 4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. — A8

SR 3.0.3 — The time limits of the Action requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The time at which the Action is taken may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the Action requirements are less than 24 hours. — M1, L3

SR 3.0.1 — Surveillance Requirements do not have to be performed on inoperable equipment. — A8

SR 3.0.4 — 4.0.4 Entry into an operational mode or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to operational modes as required to comply with Action requirements. — L1

<LATER> (5.0) — ~~4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1,2, and 3 components shall be applicable as follows:~~ — LATER

~~a. Inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i),~~

5.5.8

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the operational modes or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

LATER) (3.0) 4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval. LATER

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the Action requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The time at which the Action is taken may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the Action requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an operational mode or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to operational modes as required to comply with Action requirements.

A5

5.5.8 4.0.5 Surveillance Requirements for inservice ~~inspection and~~ testing of ASME Code Class 1,2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i), A5

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

LATER

<LATER>
(5.0)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification.

4.1 OPERATIONAL SAFETY ITEMS

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

A1

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

a. The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

A3
+ R
TRM

3.3.5
3.3.6
3.3.7

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

<LATER> (5.0)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications: — LATER

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and test activities.

d. Performance of the above inservice inspection and testing activit[ies] shall be in addition to other specified Surveillance Requirements

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification.

4.1 OPERATIONAL SAFETY ITEMS — A1

Applicability

Applies to items directly related to safety limits and limiting conditions fo[r] operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification — A3

a. The minimum frequency and type of surveillance required for react[or] protective system and engineered safeguards system instrumentatio[n] when the reactor is critical shall be stated in Table 4.1-1. — & R TBM

3.3.11
3.3.12
3.3.13

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

〈LATER〉 (5D)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications: LATER

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification.

4.1 OPERATIONAL SAFETY ITEMS A1

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

a. The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1. A3 & R TRM

3.3.8
3.3.15
3.3.16

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

LATER (5.0)

LATER

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and test activities.

d. Performance of the above inservice inspection and testing activit shall be in addition to other specified Surveillance Requirements

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification.

4.1 OPERATIONAL SAFETY ITEMS

A1

Applicability

Applies to items directly related to safety limits and limiting conditions fo operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

A3

a. The minimum frequency and type of surveillance required for reacto protective system and engineered safeguards system instrumentatio when the reactor is critical shall be as stated in Table 4.1-1.

R
TRM

5.5.8

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

5.5.8

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications: A5

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually <ADD: Every 9 mo">	At least once per 366 days
<ADD: "Biennially">	

A5

A7

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and test activities. A5

d. ~~Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements~~ A1

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification. <INSERT SR 3.0.3 appl.> A3

4.1 OPERATIONAL SAFETY ITEMS

<LATER> (3.3A) (3.3B) (3.3C) (3.3D)

~~Applicability~~

~~Applies to items directly related to safety limits and limiting conditions for operation.~~

~~Objective~~

~~To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.~~

~~Specification~~

~~a. The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.~~

LATER

Items on this page also addressed in the following packages: 3.3A, 3.3B, 3.3C, 3.3D, 5.5

SR

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification.

4.1 OPERATIONAL SAFETY ITEMS

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

R TRM

a. ~~The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.~~

3.0

ITS

<LATER>
(3.3A)
(3.3B)
(3.3C)
(3.3.D)

OPERATIONAL SAFETY ITEMS (continued)

4.1 (Continued)

b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.

c. Discrepancies noted during surveillance testing will be corrected and recorded.

LATER

<LATER>
(3.2)

d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.

LATER

BASES

4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):

"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

A2

⟨LATER⟩ (3.3A, 3.3B, 3.3C, 3.3D)

OPERATIONAL SAFETY ITEMS (continued)

4.1 (Continued)

b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.

c. Discrepancies noted during surveillance testing will be corrected and recorded.

LATER

M17

3.2.5 LCO

SR 3.2.5.1

(as specified by applicable LCO(s))

d. A power distribution ~~map shall be made to verify the expected~~ power distribution ~~at periodic intervals at least every 10 effective full power days~~ using the incore instrumentation detector system.

L11

BASES

4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):

"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

⟨LATER⟩ (3.0)

4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

LATER

⟨Add 3.2.5 Condition A⟩ M9

⟨Add 3.2.5 Condition B⟩ M9

⟨Add SR 3.2.5.1 Note⟩ L11

⟨Add 3.2.5 Applicability⟩ A7

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

~~OPERATIONAL SAFETY ITEMS (continued)~~ A1

~~4.1 (Continued)~~

& (R) TRM

~~b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.~~ A3

& <LATER> (3.3B, 3.3C, 3.3D)

~~c. Discrepancies noted during surveillance testing will be corrected and recorded.~~

<LATER> (3.2)

~~d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.~~ LATER

~~**BASES**~~ A2

~~4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):~~

~~"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."~~

~~4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.~~

3.3.5
3.3.6
3.3.7

OPERATIONAL SAFETY ITEMS (continued) (A3)

4.1 (Continued)

<LATER> (3.3A, 3.3C, 3.3D)

b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3. (R) TRM

c. Discrepancies noted during surveillance testing will be corrected and recorded. (A14)

<LATER> (3.2)

d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system. LATER

BASES (A2)

4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):

"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

3.3.11
3.3.12
3.3.13

~~OPERATIONAL SAFETY ITEMS (continued)~~ A1

~~4.1 (Continued)~~

{LATER} (3.3A, 3.3B, 3.3D)

~~b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.~~ A3 & LATER & R TRM

~~c. Discrepancies noted during surveillance testing will be corrected and recorded.~~ A17 & LATER

⟨LATER⟩ (3.2)

~~d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.~~ LATER

~~**BASES**~~ A2

~~4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):~~

~~"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."~~

~~4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.~~

3.3.8
3.3.15
3.3.16

~~OPERATIONAL SAFETY ITEMS (Continued)~~ A3 & R TRM

~~4.1 (Continued)~~

~~b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.~~

<LATER> (3.3A, 3.3B, 3.3C)

~~c. Discrepancies noted during surveillance testing will be corrected and recorded.~~ A14

<LATER> (3.2)

~~d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.~~ LATER

~~BASES~~ A2

~~4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):~~

~~"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."~~

~~4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.~~

5.5.8

OPERATIONAL SAFETY ITEMS (continued)

4.1 (Continued)

<LATER> (3.3A) (3.3B) (3.3C) (3.3D)

b. ~~Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.~~

c. ~~Discrepancies noted during surveillance testing will be corrected and recorded.~~

LATER

<LATER> (3.2)

d. ~~A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.~~

LATER

BASES

A2

~~4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):~~

~~"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."~~

~~4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.~~

Items on this page also addressed in the following packages: 3.0, 3.2, 3.3 A, 3.3B, 3.3C, 3.3D, 5.5

SR

OPERATIONAL SAFETY ITEMS (continued)

4.1 (Continued)

b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3. R TRM

c. Discrepancies noted during surveillance testing will be corrected and recorded.

d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.

BASES

4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):

"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

3.0

BASES (continued)

4.0.2 Establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance intervals.

4.0.3 Establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the Action requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the Action requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the Action requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2 was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to enforcement action. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications. A2

If the allowable outage time limits of the Action requirements are less than 24 hours or a shutdown is required to comply with Action requirements, e.g., Specification 3.0.3, a 24-hour allowance is provided to permit a delay in implementing the Action requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with Action requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance

3.0

BASES (continued)

includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of mode changes imposed by Action requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the Action requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the Action requirements are applicable at the time that the surveillance is terminated. If the Action requirements are greater than 24 hours, sufficient time exists to complete the surveillance.

Surveillance Requirements do not have to be performed on inoperable equipment because the Action requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

A2

4.0.4 Establishes the requirement that all applicable surveillances must be met before entry into an operational mode or other condition of operation specified in the Specification. The purpose of this Specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a mode or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in operational modes or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with Action requirements, the provision of Specification 4.0.4 do not apply because this would delay placing the facility in a lower mode of operation.

4.0.5 Establishes the requirment that inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

LATER

<LATER>
(5.0)

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

5.5.8

BASES (continued)

includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of mode changes imposed by Action requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the Action requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the Action requirements are applicable at the time that the surveillance is terminated. If the Action requirements are greater than 24 hours, sufficient time exists to complete the surveillance.

<LATER> (3.0) — LATER

Surveillance Requirements do not have to be performed on inoperable equipment because the Action requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

4.0.4 Establishes the requirement that all applicable surveillances must be met before entry into an operational mode or other condition of operation specified in the Specification. The purpose of this Specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a mode or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in operational modes or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with Action requirements, the provision of Specification 4.0.4 do not apply because this would delay placing the facility in a lower mode of operation.

A2

4.0.5 Establishes the requirment that inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable. A2

4.1 Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator Action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation. A2 & R TRM

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift. During nonsteady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. A2

3.3.5
3.3.6
3.3.7

BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable. A2

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3.3.11
3.3.12
3.3.13

BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable. A2

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3.3.8
3.3.15
3.3.16

BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable. A2

4.1 Bases

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5.5.8

BASES (continued)

~~Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.~~ A2

4.1 Bases

Check

~~Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator Action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.~~ A2 & R TRM

Calibration

~~Calibration shall be performed to assure the presentation and acquisition of accurate information.~~ ~~The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift.~~ ~~During nonsteady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.~~ A2

Items on this page also addressed in the following packages: 3.3A, 3.3B, 3.3C, 3.3D, 5.5

5R

BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

4.1 Bases

Check

~~Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator Action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.~~

Calibration

~~Calibration shall be performed to assure the presentation and acquisition of accurate information.~~ The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift. During nonsteady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

R
TRM

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

Other channels are subject only to "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed once every 18 months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies for the nuclear flux (power range) channels, and once every 18 months for the process system channels is considered acceptable.

Testing

On-line testing of reactor protective channel and EFIC channels is required once every 4 weeks on a rotational or staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

All reactor protective channels will be tested before startup if the individual channel rotational frequency has been discontinued or if outage activities could potentially have affected the operability of one or more channels. A rotation will then be established to test the first Channel one week after startup, the second Channel two weeks after startup, the third Channel three weeks after startup, and the fourth Channel four weeks after startup. A2

The established reactor protective system instrumentation and EFIC test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every quarter. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protective channels shall be trip tested prior to startup and their individual channels trip tested on a cyclic basis. Discovery of a failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

3.3.5
3.3.6
3.3.7

Other channels are subject only to "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed once every 18 months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies for the nuclear flux (power range) channels, and once every 18 months for the process system channels is considered acceptable.

Testing

On-line testing of reactor protective channel and EFIC channels is required once every 4 weeks on a rotational or staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel. A2

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3.3.11
3.3.12
3.3.13

Other channels are subject only to "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed once every 18 months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies for the nuclear flux (power range) channels, and once every 18 months for the process system channels is considered acceptable.

<u>Testing</u>

On-line testing of reactor protective channel and EFIC channels is required once every 4 weeks on a rotational or staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel. A2

All reactor protective channels will be tested before startup if the individual channel rotational frequency has been discontinued or if outage activities could potentially have affected the operability of one or more channels. A rotation will then be established to test the first Channel one week after startup, the second Channel two weeks after startup, the third Channel three weeks after startup, and the fourth Channel four weeks after startup.

The established reactor protective system instrumentation and EFIC test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every quarter. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protective channels shall be trip tested prior to startup and their individual channels trip tested on a cyclic basis. Discovery of a failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

3.3.8
3.3.15
3.3.16

Other channels are subject only to "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed once every 18 months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies for the nuclear flux (power range) channels, and once every 18 months for the process system channels is considered acceptable.

Testing

On-line testing of reactor protective channel and EFIC channels is required once every 4 weeks on a rotational or staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

All reactor protective channels will be tested before startup if the individual channel rotational frequency has been discontinued or if outage activities could potentially have affected the operability of one or more channels. A rotation will then be established to test the first Channel one week after startup, the second Channel two weeks after startup, the third Channel three weeks after startup, and the fourth Channel four weeks after startup. A2

The established reactor protective system instrumentation and EFIC test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every quarter. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protective channels shall be trip tested prior to startup and their individual channels trip tested on a cyclic basis. Discovery of a failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

~~The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the status of the equipment and systems to assure safe operation.~~ A2

~~**REFERENCE**~~ & R TRM

~~FSAR Section 7.1.2.3.4~~

3.3.5
3.3.6
3.3.7

A2

R
TRM

~~The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the status of the equipment and systems to assure safe operation.~~

~~**REFERENCE**~~

~~FSAR Section 7.1.2.3.4~~

3.3.11
3.3.12
3.3.13

~~The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the status of the equipment and systems to assure safe operation.~~

A2

R

TRM

~~REFERENCE~~

~~FSAR Section 7.1.2.3.4~~

3.3.8
3.3.15
3.3.16

A2

& R
TRM

~~The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the status of the equipment and systems to assure safe operation.~~

~~REFERENCE~~

~~FSAR Section 7.1.2.3.4~~

Items on this page also addressed in the following packages: 3.3 A, 3.3 B, 3.3 C, 3.3 D

SR

R TRM

~~The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the status of the equipment and systems to assure safe operation.~~

~~REFERENCE~~

~~FSAR Section 7.1.2.3.4~~

<Add SR 3.3.1.2 - NOTE> (L12)
<Add SR 3.3.1.3 - NOTE> (L13)
<Add SR 3.3.1.6 - NOTE> (A10)

(A3)

Table 4.1-1
Instrument Surveillance Requirements

	Channel Description	Check	Test	Calibrate	Remarks
SR 3.3.3.1	1. ~~Protective Channel Coincidence Logic~~ Reactor Trip Module	NA	Q	NA	
SR 3.3.4.1	2. Control Rod Drive Trip Breaker	NA	Q(1)	NA	~~(1) To include independent testing of the shunt and undervoltage trip attachments.~~
SR 3.3.1.2	3. Power Range Amplifier	NA	NA	~~T/W(1)~~ Adjust Power range Channel output if Calorimetric exceeds Power range channel by ≥ 2% RTP	(1) Heat balance calibration ~~twice weekly~~ every 96 hours ~~under steady state operating conditions~~, ~~daily~~ 24 hours under non-steady state operating conditions.
SR 3.3.1.5, SR 3.3.1.1, SR 3.3.1.3, Table 3.3.1-1 Function 1a + Function 8	4. ~~Power Range Channel~~ Nuclear Overpower and Nuclear Overpower RCS Flow and measured AXIAL POWER IMBALANCE	S M(1)	~~M~~	M(1)(2)	(1) Using core instrumentation. (2) Axial offset upper and lower chambers ~~monthly and after each startup if not done previous week.~~ every 31 days
SR 3.3.10.1, SR 3.3.10.2	5. Intermediate Range Channel	S	~~P~~ M	~~MR~~ R	
SR 3.3.9.1	6. Source Range Channel	S(1)	~~P~~	~~NA~~ R	~~(1) When in service.~~
Table 3.3.1-1 Function 2	7. Reactor Coolant Temperature Channel	S	M	R	
Function 3	8. High Reactor Coolant Pressure Channel	S	M	R	
Function 4	9. Low Reactor Coolant Pressure Channel	S	M	R	
Function 8	10. Flux-Reactor Coolant Flow Comparator	S	~~M~~	R	
Function 5	11. Reactor Coolant Pressure Temperature Comparator	S	M	R	
Function 7	12. Pump Flux Comparator	S	M	R	
		SR 3.3.1.1	SR 3.3.1.4	SR 3.3.1.6	

(A1) (LA1) Bases (A1) (M4) (L2) (A5) (L9) (A11) (A6) (A1) (M9) (L14) (L15) (A1) (A11)

Add SR 3.3.9.2 and SR 3.3.10.3 with NOTES

3.3.1
3.3.3
3.3.4
3.3.9
3.3.10

(A3)

Table 4.1-1 (cont.)

	Channel Description	Check	Test	Calibrate	Remarks
Table 3.3.1-1 Function 6	13. High Reactor Building Pressure Channel	S SR3.3.1.1	M SR3.3.1.4	R SR3.3.1.6	(A1)
	14. High Pressure Injection Logic Channel	NA	M	NA	
	15. High Pressure Injection Analog Channels				
<LATER> (3.3B, 3.3D)	a. Reactor Coolant Pressure Channel	S	M(1)	R	(1) Including test of shutdown bypass function (ECCS bypass function).
	b. Reactor Building 4 psig Channel	S	M	R	
<LATER> (3.3B)	16. Low Pressure Injection Logic Channel	NA	M	NA	
	17. Low Pressure Injection Analog Channels				
	a. Reactor Coolant Pressure Channel	S	M(1)	R	(1) Including test of shutdown bypass function (ECCS bypass function). LATER
<LATER> (3.3B, 3.3D)	b. Reactor Building 4 psig Channel	S	M	R	
	18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	M	NA	
	19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
	a. Reactor Building 4 psig Channels	S	M	R	

3.3.1

A3

Table 4.1-1 (cont.)

	Channel Description	Check	Test	Calibrate	Remarks
<LATER> (3.3A)	~~13. High Reactor Building Pressure Channel~~	~~S~~	~~M~~	~~R~~	LATER
3.3.7	14. High Pressure Injection Logic Channel	NA	M SR 3.3.7.1	NA	
3.3.5	15. High Pressure Injection Analog Channels				
<LATER> (3.3D)	a. Reactor Coolant Pressure Channel	S SR 3.3.5.1	M(1) SR 3.3.5.2	R SR 3.3.5.3	~~(1) Including test of shutdown bypass function (ECCS bypass function).~~ L7
	b. Reactor Building 4 psig Channel	S SR 3.3.5.1	M SR 3.3.5.2	R SR 3.3.5.3	
3.3.7	16. Low Pressure Injection Logic Channel	NA	M SR 3.3.7.1	NA	
3.3.5	17. Low Pressure Injection Analog Channels				
<LATER> (3.3D)	a. Reactor Coolant Pressure Channel	S SR 3.3.5.1	M(1) SR 3.3.5.2	R SR 3.3.5.3	~~(1) Including test of shutdown bypass function (ECCS bypass function).~~ L7
	b. Reactor Building 4 psig Channel	S SR 3.3.5.1	M SR 3.3.5.2	R SR 3.3.5.3	
3.3.7	18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	M SR 3.3.7.1	NA	
3.3.5	19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
	a. Reactor Building 4 psig Channels	S SR 3.3.5.1	M SR 3.3.5.2	R SR 3.3.5.3	

3.3.5
3.3.7

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.3A)	13. ~~High Reactor Building Pressure Channel~~	~~S~~	~~M~~	~~R~~		LATER
<LATER> (3.3B)	14. ~~High Pressure Injection Logic Channel~~	~~NA~~	~~M~~	~~NA~~		
	15. ~~High Pressure Injection Analog Channels~~					L10
3.3.15 PAM #4 <Later>	a. Reactor Coolant Pressure Channel	S M SR 3.3.15.1	M(1)	R SR 3.3.15.2	(1) Including test of shutdown bypass function (ECCS bypass function).	M7 & LATER
(3.3B)	b. ~~Reactor Building 4 psig Channel~~	~~S~~	~~M~~	~~R~~		LATER
<LATER> (3.3B)	16. ~~Low Pressure Injection Logic Channel~~	~~NA~~	~~M~~	~~NA~~		
	17. ~~Low Pressure Injection Analog Channels~~					L10
3.3.15 PAM #4 <Later>	a. Reactor Coolant Pressure Channel	S M SR 3.3.15.1	M(1)	R SR 3.3.15.2	(1) Including test of shutdown bypass function (ECCS bypass function).	M7 & LATER
(3.3B)	b. ~~Reactor Building 4 psig Channel~~	~~S~~	~~M~~	~~R~~		
<LATER> (3.3B)	18. ~~Reactor Building Emergency Cooling and Isolation System Logic Channel~~	~~NA~~	~~M~~	~~NA~~		LATER
	19. ~~Reactor Building Emergency Cooling and Isolation System Analog Channels~~					
	a. ~~Reactor Building 4 psig Channels~~	~~S~~	~~M~~	~~R~~		

3.3.15

Table 4.1-1 (Cont'd)

	Channel Description	Check	Test	Calibrate	Remarks	
⟨LATER⟩ (3.3 B)	20. Reactor Building Spray System System Logic Channels	NA	M(1)	NA	(1) Including RB spray pump, spray valves, and chem. add. valve logic channels.	LATER
	21. Reactor Building Spray System Analog Channels					
	a. Reactor Building Pressure Channels	NA	M	R		LATER
⟨LATER⟩ (3.3 D)	22. Pressurizer Temperature Channels	S	NA	R		
	23. Control Rod Absolute Position	S(1) SR 3.1.7.1	NA	R SR 3.1.7.2	(1) Compare with Relative Position Indicator	L 1
	24. Control Rod Relative Position	S(1) SR 3.1.7.1	NA	R SR 3.1.7.2	(1) Check with Absolute Position Indicator	
⟨LATER⟩ (3.5)	25. Core Flooding Tanks					LATER
	a. Pressure Channels	S	NA	R		
	b. Level Channels	S	NA	R		
⟨LATER⟩ (3.3 D)	26. Pressurizer Level Channels	S	NA	R		LATER
	27. Makeup Tank Level Channels	D	NA	R		
⟨LATER⟩ (3.3D & 3.4B)	28. Radiation Monitoring Systems other than containment high range monitors (item 57)				(1) Check functioning of self-checking feature on each detector.	LATER
	a. Process Monitoring System	S	Q	R		
⟨LATER⟩ (3.3D)	b. Area Monitoring System	S	M(1)	R		LATER
	c. Main Steam Line Radiation Monitors	S	M	R		

3.1.7

A3

Table 4.1-1 (Cont'd)

		Channel Description	Check	Test	Calibrate	Remarks	
3.3.7	20.	Reactor Building Spray System System Logic Channels	NA	M(1) SR 3.3.7.1	NA	(1) Including RB spray pump, spray valves, and chem. add. valve logic channels.	LA2 Bases
3.3.5	21.	Reactor Building Spray System Analog Channels	<Add SR 3.3.5.1>				A11
		a. Reactor Building Pressure Channels	NA	SR 3.3.5.2 M	SR 3.3.5.3 R		
<LATER> (3.3D)	22.	Pressurizer Temperature Channels	S	NA	R		LATER
<LATER> (3.1)	23.	Control Rod Absolute Position	S(1)	NA	R	(1) Compare with Relative Position Indicator.	LATER
	24.	Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator	
<LATER> (3.5)	25.	Core Flooding Tanks					LATER
		a. Pressure Channels	S	NA	R		
		b. Level Channels	S	NA	R		
<LATER> (3.3D)	26.	Pressurizer Level Channels	S	NA	R		LATER
<LATER> (3.4B) (3.3D)	27.	Makeup Tank Level Channels	D	NA	R		LATER
	28.	Radiation Monitoring Systems other than containment high range monitors (item 57)				(1) Check functioning of self-checking feature on each detector.	
		a. Process Monitoring System	S	Q	R		
		b. Area Monitoring System	S	M(1)	R		
(LATER) (3.3D)		c. Main Steam Line Radiation Monitors	S	M	R		LATER

Table 4.1-1 (Cont'd)

	Channel Description	Check	Test	Calibrate	Remarks	
⟨LATER⟩ (3.3B)	20. Reactor Building Spray System System Logic Channels	NA	M(1)	NA	(1) Including RB spray pump, spray valves, and chem. add. valve logic channels.	LATER
	21. Reactor Building Spray System Analog Channels					
	a. Reactor Building Pressure Channels	NA	M	R		
⟨LATER⟩ (3.3D)	22. Pressurizer Temperature Channels	S	NA	R		LATER
⟨LATER⟩ (3.1)	23. Control Rod Absolute Position	S(1)	NA	R	(1) Compare with Relative Position Indicator.	LATER
	24. Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator	
⟨LATER⟩ (3.5)	25. Core Flooding Tanks					LATER
	a. Pressure Channels	S	NA	R		
	b. Level Channels	S	NA	R		
⟨LATER⟩ (3.3D)	26. Pressurizer Level Channels	S	NA	R		LATER
	27. Makeup Tank Level Channels	D	NA	R		
	28. Radiation Monitoring Systems other than containment high range monitors (item 57)				(1) Check functioning of self-checking feature on each detector.	
3.4.15 & LATER (3.3D)	a. Process Monitoring System (RCS Leakage monitors only)	S SR3.4.15.1	Q SR3.4.15.2	R SR 3.4.15.3		& LATER
⟨LATER⟩ (3.3D)	b. Area Monitoring System	S	M(1)	R		LATER
	c. Main Steam Line Radiation Monitors	S	M	R		

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3.4.15

Table 4.1-1 (Cont'd)

	Channel Description	Check	Test	Calibrate	Remarks	
⟨LATER⟩ (3.3B)	20. Reactor Building Spray System System Logic Channels	NA	M(1)	NA	(1) Including RB spray pump, spray valves, and chem. add. valve logic channels.	LATER
	21. Reactor Building Spray System Analog Channels					
	a. Reactor Building Pressure Channels	NA	M	R		
⟨LATER⟩ (3.3D)	22. Pressurizer Temperature Channels	S	NA	R		LATER
⟨LATER⟩ (3.1)	23. Control Rod Absolute Position	S(1)	NA	R	(1) Compare with Relative Position Indicator.	LATER
	24. Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator	
	25. Core Flooding Tanks					LA3 TRM
	a. Pressure Channels	S	NA	R		
	b. Level Channels	S	NA	R		
⟨LATER⟩ (3.3D)	26. Pressurizer Level Channels	S	NA	R		LATER
	27. Makeup Tank Level Channels	D	NA	R		
⟨LATER⟩ (3.4B & 3.3D)	28. Radiation Monitoring Systems other than containment high range monitors (item 57)				(1) Check functioning of self-checking feature on each detector.	LATER
	a. Process Monitoring System	S	Q	R		
⟨LATER⟩ (3.3D)	b. Area Monitoring System	S	M(1)	R		LATER
	c. Main Steam Line Radiation Monitors	S	M	R		

3.5

(A3)

Table 4.1-1 (Cont'd)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.3B)	20. Reactor Building Spray System System Logic Channels	NA	M(1)	NA	(1) Including RB spray pump, spray valves, and chem. add. valve logic channels.	LATER
	21. Reactor Building Spray System Analog Channels					
	a. Reactor Building Pressure Channels	NA	M	R		L15
	22. Pressurizer Temperature Channels	S	NA	R		L15
<LATER> (3.1)	23. Control Rod Absolute Position	S(1)	NA	R	(1) Compare with Relative Position Indicator.	LATER
	24. Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator	
<LATER> (3.5)	25. Core Flooding Tanks					LATER
	a. Pressure Channels	S	NA	R		
	b. Level Channels	S	NA	R		L10
3.3.15 PAM #11	26. Pressurizer Level Channels	~~S~~ M SR 3.3.15.1	NA	R SR 3.3.15.2		L15
	27. Makeup Tank Level Channels	D	NA	R		LATER
<LATER> (3.4B)	28. Radiation Monitoring Systems other than containment high range monitors (item 57)					
	a. Process Monitoring System (RCS Leakage monitors only)	S	Q	R	(1) Check functioning of self-checking feature on each detector.	L13
3.3.16 Also See Page 71-2	b. Area Monitoring System (Control Room only)	S SR 3.3.16.1	M(1) SR 3.3.16.2	R SR 3.3.16.3		
	c. Main Steam Line Radiation Monitors	S	M	R		LA2 ODCM SAR

3.3.15
3.3.16

A3

Table 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
20. Reactor Building Spray System System Logic Channels	NA	M(1)	NA	(1) Including RB spray pump, spray valves, and chem. add. valve logic channels.
21. Reactor Building Spray System Analog Channels				
a. Reactor Building Pressure Channels	NA	M	R	
22. Pressurizer Temperature Channels	S	NA	R	
23. Control Rod Absolute Position	S(1)	NA	R	(1) Compare with Relative Position Indicator.
24. Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator
25. Core Flooding Tanks				
a. Pressure Channels	S	NA	R	
b. Level Channels	S	NA	R	
26. Pressurizer Level Channels	S	NA	R	
27. Makeup Tank Level Channels	D	NA	R	
~~28. Radiation Monitoring Systems other than containment high range monitors (item 57)~~				~~(1) Check functioning of self-checking feature on each detector.~~
~~a. Process Monitoring System~~ (except RCS Leakage Monitoring)	~~S~~	~~Q~~	~~R~~	
~~b. Area Monitoring System~~ (except Control Room)	~~S~~	~~M(1)~~	~~R~~	
c. Main Steam Line Radiation Monitors	S	M	R	

See Page 71-1

See Page 71-1

L15

3.3.16

(A3)

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.3 D)	29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R		LATER
<LATER> (3.4.8)	30. Decay heat removal system isolation valve automatic closure and interlock system	S(1)(2)	M(1)(3)	R	(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) At least once every refueling shutdown, with Reactor Coolant System Pressure greater than or equal to 200 psig, but less than 300 psig, verify automatic isolation of the decay heat removal system from the Reactor Coolant System on high Reactor Coolant System pressure.	LATER
	31. Deleted					A1
<LATER> (3.8)	32. Diesel generator protective relaying starting interlocks and circuitry	M	Q	NA		LATER
<LATER> (3.8)	33. Off-site power undervoltage and protective relaying interlocks and circuitry	W	R(1)	R(1)	(1) Shall be tested during refueling shutdown to demonstrate selective load shedding interlocks function during manual or automatic transfer of Unit 1 auxiliary load to Startup Transformer No. 2.	LATER
<LATER> (3.3 D)	34. Borated water storage tank level indicator	W	NA	R		LATER
Table 3.3.1-1 Function 10	35. Reactor trip upon loss of main feedwater circuitry	~~M~~ 12 hour SR 3.3.1.1	~~PC~~ 31 days SR 3.3.1.4	R SR 3.3.1.6		M2

3.3.1

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
⟨LATER⟩ (3.3 D)	29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R		LATER
3.4.14	30. Decay heat removal system isolation valve automatic closure and interlock system	S(1)(2)	M(1) (3)	R	(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) At least once every refueling shutdown, with Reactor Coolant System Pressure greater than or equal to 200 psig, but less than 300 psig, verify automatic isolation of the decay heat removal system from the Reactor Coolant System on high Reactor Coolant System pressure.	⟨SR 3.4.14.2 SR 3.4.14.4 & SR 3.4.14.5⟩ L4 ⟨SR 3.4.14.3⟩ LA2 TRM
	31. Deleted					A1
⟨LATER⟩ (3.8)	32. Diesel generator protective relaying starting interlocks and circuitry	M	Q	NA		
	33. Off-site power undervoltage and protective relaying interlocks and circuitry	W	R(1)	R(1)	(1) Shall be tested during refueling shutdown to demonstrate selective load shedding interlocks function during manual or automatic transfer of Unit 1 auxiliary load to Startup Transformer No. 2	LATER
⟨LATER⟩ (3.3 D)	34. Borated water storage tank level indicator	W	NA	R		LATER
⟨LATER⟩ (3.3 A)	35. Reactor trip upon loss of main feedwater circuitry	M	PC	R		LATER

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
3.3.15 PAMS #18 & #19	29. High and Low Pressure Injection Systems: Flow Channels	~~NA~~ M <Add SR 3.3.15.1>	NA	R SR 3.3.15.2	M7
<LATER> (3.4B)	~~30. Decay heat removal system isolation valve automatic closure and interlock system~~	~~S(1)(2)~~	~~M(1)(3)~~	~~R~~	LATER ~~(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) At least once every refueling shutdown, with Reactor Coolant System Pressure greater than or equal to 200 psig, but less than 300 psig, verify automatic isolation of the decay heat removal system from the Reactor Coolant System on high Reactor Coolant System pressure.~~
	31. ~~Deleted~~				A1
<LATER> (3.8)	~~32. Diesel generator protective relaying starting interlocks and circuitry~~	~~M~~	~~Q~~	~~NA~~	
	~~33. Off-site power undervoltage and protective relaying interlocks and circuitry~~	~~M~~	~~R(1)~~	~~R(1)~~	LATER ~~(1) Shall be tested during refueling shutdown to demonstrate selective load shedding interlocks function during manual or automatic transfer of Unit 1 auxiliary load to Startup Transformer No. 2.~~
3.3.15 PAM #15	34. Borated water storage tank level indicator	~~W~~ M SR 3.3.15.1	NA	R SR 3.3.15.2	L10
<LATER> (3.3A)	~~35. Reactor trip upon loss of main feedwater circuitry~~	~~M~~	~~PC~~	~~R~~	LATER

3.3.15

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.3 D)	29. ~~High and Low Pressure Injection Systems: Flow Channels~~	~~NA~~	~~NA~~	~~R~~		LATER
<LATER> (3.4 B)	30. ~~Decay heat removal system isolation valve automatic closure and interlock system~~	~~S(1)(2)~~	~~M(1)(3)~~	~~R~~	~~(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) At least once every refueling shutdown, with Reactor Coolant System Pressure greater than or equal to 200 psig, but less than 300 psig, verify automatic isolation of the decay heat removal system from the Reactor Coolant System on high Reactor Coolant System pressure.~~	LATER
	31. ~~Deleted~~					A1
SR 3.8.1.7 SR 3.8.1.8	32. ~~Diesel generator protective relaying starting interlocks and circuitry~~	~~M~~	~~Q~~ (R)	~~NA~~		L14 LA1 Bases
SR 3.8.1.6	33. Off-site power undervoltage and protective relaying interlocks and circuitry	~~M~~	~~R(1)~~	~~R(1)~~	~~(1) Shall be tested during refueling shutdown to demonstrate selective load shedding interlocks function during manual or automatic transfer of Unit 1 auxiliary load to Startup Transformer No. 2.~~	LA3 TRM LA1 Bases
<LATER> (3.3 D)	34. ~~Borated water storage tank level indicator~~	~~W~~	~~NA~~	~~R~~		LATER
<LATER> (3.3 A)	35. ~~Reactor trip upon loss of main feedwater circuitry~~	~~M~~	~~PC~~	~~R~~		LATER

3.8.1

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.5)	36. Boric Acid Addition Tank					LATER
	a. Level Channel	NA	NA	R		
	b. Temperature Channel	M	NA	R		
<LATER> (3.3D)	37. Degraded Voltage Monitoring	W	R	R		LATER
	38. Sodium Hydroxide Tank Level Indicator	NA	NA	R		R TRM
	39. Incore Neutron Detectors	M(1)	NA	NA	(1) Check Functioning	LA1 TRM
<LATER> (3.3D)	40. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check	LATER
<LATER> (3.3A)	41. Reactor Trip Upon Turbine Trip Circuitry	M	PC	R		LATER
	42. Deleted					A1

3.2

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.5)	~~36. Boric Acid Addition Tank~~					
	~~a. Level Channel~~	~~NA~~	~~NA~~	~~R~~		LATER
	~~b. Temperature Channel~~	~~M~~	~~NA~~	~~R~~		
<LATER> (3.3D)	~~37. Degraded Voltage Monitoring~~	~~W~~	~~R~~	~~R~~		LATER
	~~38. Sodium Hydroxide Tank Level Indicator~~	~~NA~~	~~NA~~	~~R~~		R TRM
<LATER> (3.2)	~~39. Incore Neutron Detectors~~	~~M(1)~~	~~NA~~	~~NA~~	~~(1) Check Functioning~~	LATER
<LATER> (3.3D)	~~40. Emergency Plant Radiation Instruments~~	~~M(1)~~	~~NA~~	~~R~~	~~(1) Battery Check~~	LATER
Table 3.3.1-1 Function 9	41. Reactor Trip Upon Turbine Trip Circuitry	~~M~~ 12 hours SR 3.3.1.1	~~PC~~ 31 days SR 3.3.1.4	R SR 3.3.1.6		M2
	~~42. Deleted~~					A1

3.3.1

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.5)	36. Boric Acid Addition Tank					LATER
	a. Level Channel	NA	NA	R		A5
	b. Temperature Channel	M	NA	R		
3.3.8	37. Degraded Voltage Monitoring	W SR 3.3.8.1	R	R SR 3.3.8.2	<Add SR 3.3.8.2 Note>	L2
	38. Sodium Hydroxide Tank Level Indicator	NA	NA	R		R TRM
<LATER> (3.2)	39. Incore Neutron Detectors	M(1)	NA	NA	(1) Check Functioning	LATER
	40. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check	L15
<LATER> (3.3A)	41. Reactor Trip Upon Turbine Trip Circuitry	M	PC	R		LATER
	42. Deleted					A1

3.3.8

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
	36. ~~Boric Acid Addition Tank~~					LA3 TRM
	a. ~~Level Channel~~	~~NA~~	~~NA~~	~~R~~		
	b. ~~Temperature Channel~~	~~M~~	~~NA~~	~~R~~		
<LATER> (3.3D)	37. ~~Degraded Voltage Monitoring~~	~~W~~	~~R~~	~~R~~		LATER
	38. ~~Sodium Hydroxide Tank Level Indicator~~	~~NA~~	~~NA~~	~~R~~		R TRM
<LATER> (3.2)	39. ~~Incore Neutron Detectors~~	~~M(1)~~	~~NA~~	~~NA~~	~~(1) Check Functioning~~	LATER
<LATER> (3.3D)	40. ~~Emergency Plant Radiation Instruments~~	~~M(1)~~	~~NA~~	~~R~~	~~(1) Battery Check~~	LATER
<LATER> (3.3A)	41. ~~Reactor Trip Upon Turbine Trip Circuitry~~	~~M~~	~~PC~~	~~R~~		LATER
	42. ~~Deleted~~					A1

3.5

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
36. Boric Acid Addition Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	
37. Degraded Voltage Monitoring	W	R	R	
~~38. Sodium Hydroxide Tank Level Indicator~~	~~NA~~	~~NA~~	~~R~~	R TRM
39. Incore Neutron Detectors	M(1)	NA	NA	(1) Check Functioning
40. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check
41. Reactor Trip Upon Turbine Trip Circuitry	M	PC	R	
42. Deleted				

Items on this page also addressed in the following packages: 3.2, 3.3A, 3.3D, 3.5

SR

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
<LATER> (3.3B)	~~43. ESAS Manual Trip Functions~~				LATER
	~~a. Switches & Logic~~	~~NA~~	~~R~~	~~NA~~	
	~~b. Logic~~	~~NA~~	~~M~~	~~NA~~	
SR 3.3.2.1	44. Reactor Manual Trip	NA	P	NA	A1
<LATER> (3.4B)	~~45. Reactor Building Sump Level~~	~~NA~~	~~NA~~	~~R~~	LATER
<LATER> (3.3D)	~~46. EFW Flow Indication~~	~~M~~	~~NA~~	~~R~~	LATER

3.3.2

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A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
	43. ESAS Manual Trip Functions					A1
3.3.6	a. Switches & Logic	NA	SR3.3.6.1 R	NA		
3.3.7	b. Logic	NA	M SR3.3.7.1	NA		A10
<LATER> (3.3A)	~~44. Reactor Manual Trip~~	~~NA~~	~~P~~	~~NA~~	LATER	
<LATER> (3.4B)	~~45. Reactor Building Sump Level~~	~~NA~~	~~NA~~	~~R~~	LATER	
<LATER> (3.3D)	~~46. EFW Flow Indication~~	~~M~~	~~NA~~	~~R~~	LATER	

3.3.6
3.3.7

A3

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
<LATER> (3.3B) ~~43. ESAS Manual Trip Functions~~				LATER
~~a. Switches & Logic~~	~~NA~~	~~R~~	~~NA~~	
~~b. Logic~~	~~NA~~	~~M~~	~~NA~~	
<LATER> (3.3A) ~~44. Reactor Manual Trip~~	~~NA~~	~~P~~	~~NA~~	LATER
<LATER> (3.4B) ~~45. Reactor Building Sump Level~~	~~NA~~	~~NA~~	~~R~~	LATER
3.3.15 PAM #17 46. EFW Flow Indication	M SR 3.3.15.1	NA	R SR 3.3.15.2	L10

3.3.15

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
⟨LATER⟩ (3.3 B)	~~43. ESAS Manual Trip Functions~~				
	~~a. Switches & Logic~~	~~NA~~	~~R~~	~~NA~~	LATER
	~~b. Logic~~	~~NA~~	~~M~~	~~NA~~	
⟨LATER⟩ (3.3 A)	~~44. Reactor Manual Trip~~	~~NA~~	~~P~~	~~NA~~	LATER
3.4.15	45. Reactor Building Sump Level	NA	NA	R SR 3.4.15.4	
⟨LATER⟩ (3.3 D)	~~46. EFW Flow Indication~~	~~M~~	~~NA~~	~~R~~	LATER

3.4.15

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
	47. RCS Subcooling Margin Monitor	D	NA	R		L15
	48. Electromatic Relief Valve Flow Monitor	D	NA	R		L15
	49. Electromatic Relief Block Valve Position Indicator	D	NA	R		L15
	50. Pressurizer Safety Valve Flow Monitor	D	NA	R		L15
3.3.15 PAM #11	51. Pressurizer Water Level Indicator	~~D~~ M SR 3.3.15.1	NA	R SR 3.3.15.2		L10
	52. Deleted					A1
<LATER> (3.3C)	53. EFW Initiation					
	a. Manual	NA	M	NA		LATER
3.3.15 PAMS 12 & 13	b. SG Low Level, SGA or B	~~S~~ M SR 3.3.15.1	M & LATER	R SR 3.3.15.2		& LATER; L10
& <LATER> (3.3C)	c. Low Pressure SGA or B	~~S~~ M	M	R		& LATER; L10
<LATER> (3.3C)	d. Loss of both MFW Pumps and PWR > 10%	S	M	R		LATER

3.3.15

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
⟨LATER⟩ (3.3 D)	47. RCS Subcooling Margin Monitor	D	NA	R	LATER
	48. Electromatic Relief Valve Flow Monitor	D	NA	R	
	49. Electromatic Relief Block Valve Position Indicator	D	NA	R	
	50. Pressurizer Safety Valve Flow Monitor	D	NA	R	
	51. Pressurizer Water Level Indicator	D	NA	R	
	52. Deleted				A1
3.3.12	53. EFW Initiation				
Table 3.3.11-1	a. Manual	NA	M- SR 3.3.12.1	NA	
#1.b	b. SG Low Level, SGA or B	S	M	R	≶ LATER
#1.C	c. Low Pressure SGA or B	S	M	R	≶ LATER
#1.a	d. Loss of both MFW Pumps and PWR > 10%	S	M	R	A14
≶⟨LATER⟩ (3.3D)		SR 3.3.11.1	SR 3.3.11.2	SR 3.3.11.3	

3.3.11
3.3.12

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
T3.3.11-1 #1.d	e. Loss of 4 RC pumps	S-SR3.3.11.1	M-SR3.3.11.2	NA	
3.3.13	f. ESAS automatic logic tripped	NA	M-SR3.3.13.1	NA	
	54. SGA main steam line isolation				
3.3.12	a. Manual	NA	M-SR3.3.12.1	NA	
T3.3.11-1 #3a	b. SGA pressure low	S-SR3.3.11.1	M-SR3.3.11.2	R-SR3.3.11.3	
	55. SGB main steam line isolation				
3.3.12	a. Manual	NA	M-SR3.3.12.1	NA	
T3.3.11-1 #3a	b. SGB pressure low	S-SR3.3.11.1	M-SR3.3.11.2	R-SR3.3.11.3	
	56. EFW valve commands (Vector)				
T3.3.11-1 #2.a	a. SG A pressure low	S	M	R	
#2.a	b. SG B pressure low	S	M	R	
#2.b	c. SG pressure difference SG A pressure> SG B pressure	S	M	R	
#2b	SG B pressure> SG A pressure	S	M	R	
		SR 3.3.11.1	SR 3.3.11.2	SR 3.3.11.3	

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3.3.11
3.3.12
3.3.13

(A3)

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate		Remarks
<LATER> (3.3c)	d. SG A High Range Level High-high	S	M	R	LATER	
	e. SG B High Range Level High-high	S	M	R		
<LATER> (3.3D)	57. Containment High Range Radiation Monitors	D	M	R	LATER	
	58. Containment Pressure-High	M	NA	R		
	59. Containment Water Level-Wide Range	M	NA	R		
<LATER> (3.4B)	60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	LATER	
<LATER> (3.3D)	61. Core-exit Thermocouples	M	NA	R	LATER	
SR 3.3.4.1	62 Electronic (SCR) Trip Relays	NA	Q	NA	(A1)	
<LATER> (3.3D)	63 RVLMS	M	NA	R	LATER	
	64 HLLMS	M	NA	R		

<LATER> (3.3B, 3.3C, 3.3D, 3.4B)

NOTE:

S - Each Shift
W - Weekly
M - Monthly
D - Daily

T/W - Twice per Week
Q - Quarterly
P - Prior to each startup if not done previous week
B/M - Every 2 months

R - Once every 18 months
PC - Prior to going Critical if not done within previous 31 days
NA - Not Applicable
SA - SA Twice per Year

(A4) + LATER + (R) TRM

Amendment No. ~~91~~, ~~94~~, ~~95~~, ~~116~~, ~~117~~, ~~135~~, ~~151~~, 194

3.3.4

A3

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
d. SG A High Range Level High-high	S	M	R	
e. SG B High Range Level High-high	S	M	R	
57. Containment High Range Radiation Monitors	D	M	R	
58. Containment Pressure-High	M	NA	R	
59. Containment Water Level-Wide Range	M	NA	R	
60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	
61. Core-exit Thermocouples	M	NA	R	
62 Electronic (SCR) Trip Relays	NA	Q	NA	
63 RVLMS	M	NA	R	
64 HDLMS	M	NA	R	

<LATER> (3.3C)

LATER

<LATER> (3.3D)

LATER

<LATER> (3.4B)

<LATER> (3.3D)

<LATER> (3.3A)

<LATER> (3.3D)

NOTE:

S - Each Shift
W - Weekly
M - Monthly
D - Daily

T/W - Twice per Week
Q - Quarterly
P - Prior to each startup if not done previous week
B/M - Every 2 months

R - Once every 18 months
PC - Prior to going Critical if not done within previous 31 days
NA - Not Applicable
SA - SA Twice per Year

+ <LATER> (3.3A, 3.3C, 3.3D, 3.4B)

A4 +LATER + R TRM

3.3.5 3.3.6 3.3.7

(A3)

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
	d. SG A High Range Level High-high	S	M	R	A10
	e. SG B High Range Level High-high	S	M	R	
<LATER> (3.3D)	57. Containment High Range Radiation Monitors	D	M	R	LATER
	58. Containment Pressure-High	M	NA	R	
	59. Containment Water Level-Wide Range	M	NA	R	
<LATER> (3.4B)	60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	LATER
<LATER> (3.3D)	61. Core-exit Thermocouples	M	NA	R	LATER
<LATER> (3.3A)	62. Electronic (SCR) Trip Relays	NA	Q	NA	LATER
<LATER> (3.3D)	63. RVLMS	M	NA	R	LATER
	64. HLLMS	M	NA	R	

+<LATER> (3.3A, 3.3B, 3.3D, 3.4B)

NOTE:

S - Each Shift
W - Weekly
M - Monthly
D - Daily

T/W - Twice per Week
Q - Quarterly
P - Prior to each startup if not done previous week
B/M - Every 2 months

R - Once every 18 months
PC - Prior to going Critical if not done within previous 31 days
NA - Not Applicable
SA - Twice per Year

(A4) + LATER + (R) TRM

Amendment No. ~~91~~, ~~94~~, ~~95~~, ~~116~~, ~~117~~, ~~135~~, ~~161~~, 194

3.3.11
3.3.12
3.3.13

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
<LATER> (3.3C)	d. ~~SG A High Range Level High-high~~	~~S~~	~~M~~	~~R~~	LATER
	e. ~~SG B High Range Level High-high~~	~~S~~	~~M~~	~~R~~	
3.3.15 PAM #9	57. Containment High Range Radiation Monitors	~~D~~ M SR 3.3.15.1	~~M~~	R SR 3.3.15.2	L12 L10
3.3.15 PAM #7	58. Containment Pressure-High	M SR 3.3.15.1	NA	R SR 3.3.15.2	
3.3.15 PAM #6	59. Containment Water Level-Wide Range	M SR 3.3.15.1	NA	R SR 3.3.15.2	
<LATER> (3.4B)	60. ~~Low Temperature Overpressure Protection Alarm Logic~~	~~NA~~	~~R~~	~~R~~	LATER
3.3.15 PAM #16	61. Core-exit Thermocouples	M SR 3.3.15.1	NA	R SR 3.3.15.2	L10
<LATER> (3.3A)	62. ~~Electronic (SCR) Trip Relays~~	~~NA~~	~~Q~~	~~NA~~	LATER
3.3.15 PAM #5	63 RVLMS	M SR 3.3.15.1	NA	R SR 3.3.15.2	L10
3.3.15 PAM #3	64 HLLMS	M SR 3.3.15.1	NA	R SR 3.3.15.2	

+ <LATER> (3.3A, 3.3B, 3.3C, 3.4B)

NOTE:

~~S - Each Shift~~
~~W - Weekly~~
~~M - Monthly~~
~~D - Daily~~
~~T/W - Twice per Week~~
~~Q - Quarterly~~
~~P - Prior to each startup if not done previous week~~
~~B/M - Every 2 months~~
~~R - Once every 18 months~~
~~PC - Prior to going critical if not done within previous 31 days~~
~~NA - Not Applicable~~
~~SA - SA Twice per Year~~

A4 + LATER + R TRM

3.3.15

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
<LATER> (3.3c)	d. SG A High Range Level High-high	S	M	R	LATER
	e. SG B High Range Level High-high	S	M	R	
<LATER> (3.3D)	57. Containment High Range Radiation Monitors	D	M	R	LATER
	58. Containment Pressure-High	M	NA	R	
	59. Containment Water Level-Wide Range	M	NA	R	
	60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	LA2 TRM
<LATER> (3.3D)	61. Core-exit Thermocouples	M	NA	R	LATER
<LATER> (3.3A)	62 Electronic (SCR) Trip Relays	NA	Q	NA	LATER
<LATER> (3.3D)	63 RVLMS	M	NA	R	LATER
	64 HLLMS	M	NA	R	

✓<LATER> (3.3A 3.3B 3.3C 3.3D)

NOTE:

S - Each Shift
W - Weekly
M - Monthly
D - Daily

T/W - Twice per Week
Q - Quarterly
P - Prior to each startup if not done previous week
B/M - Every 2 months

R - Once every 18 months
PC - Prior to going Critical if not done within previous 31 days
NA - Not Applicable
SA - SA Twice per Year

A13
✓LATER
✓R
TRM

3.4.11

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
d. SG A High Range Level High-high	S	M	R	
e. SG B High Range Level High-high	S	M	R	
57. Containment High Range Radiation Monitors	D	M	R	
58. Containment Pressure-High	M	NA	R	
59. Containment Water Level-Wide Range	M	NA	R	
60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	
61. Core-exit Thermocouples	M	NA	R	
62 Electronic (SCR) Trip Relays	NA	M	NA	
63 RVLMS	M	NA	R	
64 HLLMS	M	NA	R	

NOTE:

~~S - Each Shift~~
~~W - Weekly~~
~~M - Monthly~~
~~D - Daily~~

~~T/W - Twice per Week~~
~~Q - Quarterly~~
~~P - Prior to each startup if not done previous week~~
~~B/M - Every 2 months~~

~~R - Once every 18 months~~
~~PC - Prior to going Critical if not done within previous 31 days~~
~~NA - Not Applicable~~
~~SA - SA Twice per Year~~

R TRM

Items on this page also addressed in the following packages: 3.3A, 3.3B, 3.3C, 3.3D, 3.4B

SR

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3.1.4
3.1.5
3.1.6

<Add SR 3.1.4.1> M11
<Add SR 3.1.5.1> M11
<Add SR 3.1.6.1> M11

Table 4.1-2
Minimum Equipment Test Frequency A1

Item	Test	Frequency
SR 3.1.4.3 1. Control Rods CAP	Rod Drop Times of all Full Length Rods 1/	~~Each Refueling Shutdown~~ Following Reactor Vessel Head Removal M8
SR 3.1.4.2 2. Control Rod CAP Movement	Movement of Each CONTROL ROD ~~Rod~~	Every ~~two weeks~~ 92 days ~~Above Cold Shutdown Conditions~~ In MODES 1 and 2. L1 L3 L12
<LATER> (3.4B) 3. Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Month LATER
<LATER> (3.7) 4. Main Steam Safety Valves	Setpoint	Four Valves Every 18 Mon LATER
5. Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown R TRM
<LATER> (3.4B) 6a. Reactor Coolant System Leakage	Evaluate	Daily
b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2 LATER
7. Emergency-powered Pressurizer Heaters	Power availability	Daily
	Heater capacity functional test	Every 18 Months
<LATER> (3.6) 8. Reactor Building Isolation Trip	Functioning	Every 18 Months LATER
<LATER> (3.7) 9. Service Water Systems	Functioning	Every 18 Months LATER
10. Spent Fuel Cooling System	Functioning	Every 18 Months when irradiated fuel is in the pool R TRM

1/ Same as tests listed in Section 4.7 A1

<LATER> (3.4B)

Notes:

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement. LATER

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

3.4.9
3.4.10
3.4.13
3.4.14

Table 4.1-2
Minimum Equipment Test Frequency

	Item	Test	Frequency	
(LATER) (3.1)	~~1. Control Rods~~	~~Rod Drop Times of all Full Length Rods 1/~~	~~Each Refueling Shutdown~~	LATER
	~~2. Control Rod Movement~~	~~Movement of Each Rod~~	~~Every Two Weeks Above Cold Shutdown Conditions~~	
SR 3.4.10.1	3. Pressurizer Code Safety Valves	~~Setpoint~~	~~One Valve Every 18 Month~~	LA3 IST
(LATER) (3.7)	~~4. Main Steam Safety Valves~~	~~Setpoint~~	~~Four Valves Every 18 Months~~	LATER
	~~5. Refueling System Interlocks~~	~~Functioning~~	~~Start of Each Refueling Shutdown~~	R TRM
SR 3.4.13.1	6a. Reactor Coolant System Leakage	Evaluate <Add SR 3.4.13.1 Note>	Daily	L12 M1
SR 3.4.14.1	b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2	A1 L16
	7. Emergency-powered Pressurizer Heaters	~~Power availability~~	~~Daily~~	LA2 TRM
SR 3.4.9.2		Heater capacity functional test	Every 18 Months	
(LATER) (3.6)	~~8. Reactor Building Isolation Trip~~	~~Functioning~~	~~Every 18 Months~~	LATER
(LATER) (3.7)	~~9. Service Water Systems~~	~~Functioning~~	~~Every 18 Months~~	LATER
	~~10. Spent Fuel Cooling System~~	~~Functioning~~	~~Every 18 Months when irradiated fuel is in the pool~~	R TRM

(LATER) (3.1) ~~1/ Same as tests listed in Section 4.7~~ LATER

LA3 IST

Notes:

SR 3.4.14.1 (1) Leak testing for each valve shall be individually accomplished to demonstrate operability ~~following each refueling~~, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, ~~and prior to returning the valve to service after maintenance, repair or replacement.~~ L1 LA3 IST

~~(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.~~ LA2 TRM

3.6.3

Table 4.1-2
Minimum Equipment Test Frequency

Item	Test	Frequency	
⟨Later⟩ (3.1) 1. Control Rods	Rod Drop Times of all Full Length Rods 1/	Each Refueling Shutdown	LATER
2. Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions	
⟨Later⟩ (3.4B) 3. Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Month	LATER
⟨Later⟩ (3.7) 4. Main Steam Safety Valves	Setpoint	Four Valves Every 18 Mon	LATER
5. Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown	R TRM
⟨Later⟩ (3.4B) 6a. Reactor Coolant System Leakage	Evaluate	Daily	LATER
b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2	
7. Emergency-powered Pressurizer Heaters	Power availability	Daily	
	Heater capacity functional test	Every 18 Months	A16
SR 3.6.3.5 8. Reactor Building Isolation Trip	~~Functioning~~ Verify auto. RB iso. valves actuate on act. or sim. signal	Every 18 Months	
⟨Later⟩ (3.7) 9. Service Water Systems	Functioning	Every 18 Months	LATER
10. Spent Fuel Cooling System	Functioning	Every 18 Months when irradiated fuel is in the pool	R TRM

⟨Later⟩ (3.1) 1/ Same as tests listed in Section 4.7 — Later

Notes:

⟨Later⟩ (3.4B) — LATER

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement.

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

3.7.1
3.7.7

Table 4.1-2
Minimum Equipment Test Frequency

	Item	Test	Frequency	
<LATER> (3.1)	1. Control Rods	Rod Drop Times of all Full Length Rods 1/	Each Refueling Shutdown	LATER
	2. Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions	
<LATER> (3.4B)	3. Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Month	LATER
SR 3.7.1.1	4. Main Steam Safety Valves <Add SR 3.7.1.1, Note>	Setpoint	~~Four Valves Every 18 Mon~~	LA2 IST L2
	5. Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown	R TRM
<LATER> (3.4B)	6a. Reactor Coolant System Leakage	Evaluate	Daily	LATER
	b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2	
	7. Emergency-powered Pressurizer Heaters	Power availability	Daily	
		Heater capacity functional test	Every 18 Months	
<LATER> (3.6)	8. Reactor Building Isolation Trip	Functioning	Every 18 Months	LATER
SR 3.7.7.2	9. Service Water Systems	Functioning	Every 18 Months	M12
	10. Spent Fuel Cooling System	Functioning	Every 18 Months when irradiated fuel is in the pool	R TRM

<LATER> (3.1) 1/ Same as tests listed in Section 4.7 LATER

Notes:

<LATER> (3.4B) LATER

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement.

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

Items on this page also addressed in the following packages: 3.1, 3.4B, 3.6, 3.7

SR

Table 4.1-2
Minimum Equipment Test Frequency

Item	Test	Frequency
1. Control Rods	Rod Drop Times of all Full Length Rods 1/	Each Refueling Shutdown
2. Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions
3. Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Months
4. Main Steam Safety Valves	Setpoint	Four Valves Every 18 Months
~~5. Refueling System Interlocks~~	~~Functioning~~	~~Start of Each Refueling Shutdown~~ R TRM
6a. Reactor Coolant System Leakage	Evaluate	Daily
b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2
7. Emergency-powered Pressurizer Heaters	Power availability	Daily
	Heater capacity functional test	Every 18 Months
8. Reactor Building Isolation Trip	Functioning	Every 18 Months
9. Service Water Systems	Functioning	Every 18 Months
~~10. Spent Fuel Cooling System~~	~~Functioning~~	~~Every 18 Months when irradiated fuel is in the pool~~ R TRM

1/ Same as tests listed in Section 4.7

Notes:

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement.

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

3.4A

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

	Item	Test	Frequency	
⟨LATER⟩ (3.4B)	11. Decay heat removal system isolation valve automatic closure and isolation system	Functioning	Each Refueling Shutdown	LATER
⟨LATER⟩ (5.0)	12. Flow limiting annulus on main feedwater line at reactor building penetration	Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.	One year, two years, three years, and every five years thereafter measured from date of initial test.	LATER
⟨LATER⟩ (3.7)	13. Main steam isolation valves	a. Exercise through approximately 10% travel	a. Quarterly	LATER
		b. Cycle	b. Every 18 months	
	14. Main feedwater isolation valves	a. Exercise through approximately 5% travel	a. Quarterly	
		b. Cycle	b. Every 18 months	
	15. Reactor internals vent valves	Demonstrate operability by: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs (applied vertically upward).	Each refueling shutdown.	LA3 TRM

3.4.14

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

	Item	Test	Frequency	
SR 3.4.14.2 SR 3.4.14.3 SR 3.4.14.4 SR 3.4.14.5	11. Decay heat removal system isolation valve automatic closure and isolation system	Functioning	~~Each Refueling Shutdown~~ 18 months	L9 M4
⟨LATER⟩ (5.0)	~~12. Flow limiting annulus on main feedwater line at reactor building penetration~~	~~Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.~~	~~One year, two years, three years, and every five years thereafter measured from date of initial test.~~	Later
	~~13. Main steam isolation valves~~	~~a. Exercise through approximately 10% travel~~ ~~b. Cycle~~	~~a. Quarterly~~ ~~b. Every 18 months~~	Later
⟨LATER⟩ (3.7)	~~14. Main feedwater isolation valves~~	~~a. Exercise through approximately 5% travel~~ ~~b. Cycle~~	~~a. Quarterly~~ ~~b. Every 18 months~~	
⟨LATER⟩ (3.4A)	~~15. Reactor internals vent valves~~	~~Demonstrate operability by:~~ ~~a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities.~~ ~~b. Verifying that the valve is not stuck in an open position, and~~ ~~c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs (applied vertically upward).~~	~~Each refueling shutdown~~	LATER

<Add SR 3.7.2.2 & SR 3.7.3.2 with Note 1 with Note 1> M5

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

	Item	Test	Frequency	
<LATER> (3.4B)	~~11. Decay heat removal system isolation valve automatic closure and isolation system~~	~~Functioning~~	~~Each Refueling Shutdown~~	LATER
<LATER> (5.0)	~~12. Flow limiting annulus on main feedwater line at reactor building penetration~~	~~Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.~~	~~One year, two years, three years, and every five years thereafter measured from date of initial test.~~	LATER
SR 3.7.2.1	13. Main steam isolation valves <Add Note to SR 3.7.2.1>	~~a. Exercise through approximately 10% travel~~	~~a. Quarterly~~	L5
		b. Cycle	~~b. Every 18 months~~	M5, LA2 IST
SR 3.7.3.1	14. Main feedwater isolation valves <Add Note for SR 3.7.3.1>	~~a. Exercise through approximately 8% travel~~	~~a. Quarterly~~	M5, L5
		b. Cycle	~~b. Every 18 months~~	LA2 IST, M5
<LATER> (3.4A)	~~15. Reactor internals vent valves~~	~~Demonstrate operability by: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs (applied vertically upward).~~	~~Each refueling shutdown~~	LATER

5.0

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

	Item	Test	Frequency	
<LATER> (3.4B)	11. Decay heat removal system isolation valve automatic closure and isolation system	Functioning	Each Refueling Shutdown	LATER
	12. Flow limiting annulus on main feedwater line at reactor building penetration	Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.	One year, two years, three years, and every five years thereafter measured from date of initial test.	L5
<LATER> (3.7)	13. Main steam isolation valves	a. Exercise through approximately 10% travel b. Cycle	a. Quarterly b. Every 18 months	LATER
	14. Main feedwater isolation valves	a. Exercise through approximately 5% travel b. Cycle	a. Quarterly b. Every 18 months	
<LATER> (3.4A)	15. Reactor internals vent valves	Demonstrate operability by: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs (applied vertically upward).	Each refueling shutdown	LATER

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3.4A

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

Item	Test	Frequency	
16. ~~RCS Vent Paths~~	~~Demonstrate operability by flow verification~~	~~At least once per 18 months during cold shutdown~~	LA2 TRM
<LATER> (3.4B) 17. ~~PORV~~	~~Exercise~~	~~End of each refueling outage~~	LATER

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

	Item	Test	Frequency	
<LATER> (3.4A)	~~16. RCS Vent Paths~~	~~Demonstrate operability by flow verification~~	~~At least once per 18 months during cold shutdown~~	LATER
SR 3.4.11.5	17. PORV	Exercise	~~End of each refueling outage~~	L9
			18 months	M4

<Add SR 3.4.11.5 Note> A12

3.1

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(7)
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)
	d. Dissolved Gases	d. Weekly (7)
	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)
	f. Boron Concentration	f. 3 times/week
	g. Radiochemical Analysis for Ē Determination (2) (4)	g. Monthly (7)
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10)
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)
6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup

⟨LATER⟩ (3.4B) ⟨LATER⟩ (3.4A) ⟨LATER⟩ (3.9) ⟨LATER⟩ (3.4B) ⟨LATER⟩ (3.5) ⟨LATER⟩ (3.7) ⟨LATER⟩ (3.6)

LATER LAI TRM LATER LATER & R TRM LATER LATER LATER R TRM LATER LATER

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

⟨LATER⟩ (3.4B) LATER

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)
	d. Dissolved Gases	d. Weekly (7)
	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)
	f. Boron Concentration	f. 3 times/week
	g. Radiochemical Analysis for $\bar{E}$ Determination (2) (4)	g. Monthly (7)
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)
6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup

<LATER> (3.48) — LATER

<LATER> (3.1) — LATER

LA2

TRM LATER & R TRM

<LATER> (3.9)

<LATER> (3.48) — LATER

<LATER> (3.5) — LATER

LATER

<LATER> (3.7)

R TRM

<LATER> (3.6) — LATER

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever t gross radioactivity concentration exceeds 10% of the limit specified i the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

<LATER> (3.48) — LATER

3.4.12

<Add SR 3.4.12.2 NOTE> L11

<Add SR 3.4.12.3 NOTE> L11

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency	
1. Reactor Coolant Samples	a. ~~Gamma Isotopic Analysis~~	~~a. Bi-weekly (7)~~	LA2 TRM
SR 3.4.12.1	b. Gross Activity Determination	b. ~~3 times/week and at least every third day (1)(6)(7)~~	L11
SR 3.4.12.2	c. Gross Radioiodine Determination	c. ~~Weekly (3)(6)(7)~~	L11
<LATER> (3.1)	d. ~~Dissolved Gases~~	~~d. Weekly (7)~~	LATER
<LATER> (3.4A)	~~e. Chemistry (Cl, F, and O_2)~~	e. 3 times/week (8)	LATER
<LATER> (3.9)	~~f. Boron Concentration~~	~~f. 3 times/week~~	LATER & R TRM
SR 3.4.12.3	g. Radiochemical Analysis for $\bar{E}$ Determination (2)(4)	g. ~~Monthly (7)~~	L11
<LATER> (3.5) ~~2. Borated Water Storage Tank Water Sample~~	~~Boron Concentration~~	~~Weekly and after each makeup~~	LATER
~~3. Core Flooding Tank Sample~~	~~Boron Concentration~~	~~Monthly and after each makeup~~	
<LATER> (3.7) ~~4. Spent Fuel Pool Water Sample~~	~~Boron Concentration~~	~~Monthly and after each makeup (9)~~	LATER
~~5. Secondary Coolant Samples~~	~~a. Gross Radioiodine Concentration~~	~~a. Weekly (5)(7)(10)~~	R TRM
	~~b. Isotopic Radioiodine Concentration (4)~~	~~b. Monthly (7)(10)~~	LATER
<LATER> (3.6) ~~6. Sodium Hydroxide Tank Sample~~	~~Sodium Hydroxide Concentration~~	~~Quarterly and after each makeup~~	LATER

Notes:

(1) ~~A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled.~~ LA1 Bases ~~Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 μCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.~~ L11

3.5.1
3.5.4

<Add SR 3.5.1.4 Frequency Note> A10

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

	Item	Test	Frequency	
	1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)	LATER
<LATER> (3.4B)		b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(7)	
		c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	
<LATER> (3.1)		d. Dissolved Gases	d. Weekly (7)	LATER
<LATER> (3.4A)		e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)	LATER
<LATER> (3.9)		f. Boron Concentration	f. 3 times/week	R TRM & LATER
<LATER> (3.4B)		g. Radiochemical Analysis for Ē Determination (2) (4)	g. Monthly (7)	LATER
SR 3.5.4.3	2. Borated Water Storage Tank Water Sample	Boron Concentration	7 days Weekly and after each makeup	L5
SR 3.5.1.4	3. Core Flooding Tank Sample	Boron Concentration	31 days Monthly and after each makeup within 12 hours of ≥ 0.2 feet that is not the result of addition from a borated water source of known concentration ≥ 2270 ppm.	L3 M3
<LATER> (3.7)	4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)	
	5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10)	
		b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)	R TRM LATER
<LATER> (3.6)	6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup	LATER

<LATER> (3.4B)

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established. LATER

3.6.6

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency	
⟨Later⟩ (3.4B) 1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)	LATER
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(	
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	
⟨LATER⟩ (3.1)	d. Dissolved Gases	d. Weekly (7)	LATER
⟨LATER⟩ (3.4A)	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)	LATER
⟨Later⟩ (3.9)	f. Boron Concentration	f. 3 times/week	LATER & (R) TRM
⟨Later⟩ (3.4B)	g. Radiochemical Analysis for $\bar{E}$ Determination (2) (4)	g. Monthly (7)	LATER
⟨Later⟩ (3.5) 2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup	LATER
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup	
⟨Later⟩ (3.7) 4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)	LATER
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10	(R) TRM
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)	M 7
SR 3.6.6.3 6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	~~Quarterly and after each makeup~~ every 184 days	L 9

Notes:

⟨Later⟩ (3.4B)

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever t gross radioactivity concentration exceeds 10% of the limit specified i the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

3.7.4
3.7.13

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

	Item	Test	Frequency	
<LATER> (3.4B)	1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)	LATER
		b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(	
<LATER> (3.1)		c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	
<LATER> (3.4A)		d. Dissolved Gases	d. Weekly (7)	LATER
		e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)	LATER
<LATER> (3.9)		f. Boron Concentration	f. 3 times/week	LATER & R TRM
<LATER> (3.4B)		g. Radiochemical Analysis for $\bar{E}$ Determination (2) (4)	g. Monthly (7)	LATER
<LATER> (3.5)	2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup	LATER
	3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup	M16
SR 3.7.13.1	4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)	LA3 TRM
SR 3.7.4.1	5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10	R TRM
		b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)	
<LATER> (3.6)	6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup	LATER

<LATER> (3.4B) LATER

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units μCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever t gross radioactivity concentration exceeds 10% of the limit specified i the Specification 3.1.4.1 or increases by 10 μCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

3.9.1

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)
	d. Dissolved Gases	d. Weekly (7)
	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)
	f. Boron Concentration	f. 3 times/week 72 hours
	g. Radiochemical Analysis for Ē Determination (2) (4)	g. Monthly (7)
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10)
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)
6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup

LATER (3.4B)

⟨LATER⟩ (3.1)

⟨LATER⟩ (3.4A)

SR 3.9.1.1

LATER (3.4B)

⟨LATER⟩ (3.5)

⟨LATER⟩ (3.7)

⟨LATER⟩ (3.6)

⟨LATER⟩ (3.4B)

LATER

LATER

LATER

M12

L2

R TRM

LATER

LATER

LATER

R TRM

LATER

LATER

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

Items on this page also addressed in the following packages: 3.1, 3.4A, 3.4B, 3.5, 3.6, 3.7, 3.9

SR

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(7)
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)
	d. Dissolved Gases	d. Weekly (7)
	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)
	f. ~~Boron Concentration~~	f. ~~3 times/week~~ (R) TRM
	g. Radiochemical Analysis for Ē Determination (2) (4)	g. Montr`. (7)
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)
5. ~~Secondary Coolant Samples~~	a. ~~Gross Radioiodine Concentration~~	a. ~~Weekly (5)(7)(10)~~ (R) TRM
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)
6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 μCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

⟨LATER⟩ (3.4B)

(2) A radiochemical analysis shall consist of the quantitative measurement the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of Ē. A radiochemical analysis and calculation of Ē and iodine isotopic activity shall be performed if the measured gross activity changes by more than µCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) and beta energy per disintegrat shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes.

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 µCi/gm from the previous measured level.

LATER

⟨LATER⟩ (3.4B, 3.7)

(4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity.

LATER

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2.

R TRM

⟨LATER⟩ (3.4B)

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less that 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above.

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prior to any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above.

LATER

R TRM & LATER

& ⟨LATER⟩ (3.4B, 3.7)

(7) Not required when plant is in the cold shutdown condition or refueling shutdown condition.

LA1 TRM

⟨LATER⟩ (3.4A)

(8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition.

LATER

⟨LATER⟩ (3.7)

(9) Required only when fuel is in the pool and prior to transferring fuel the pool.

LATER

& ⟨LATER⟩ (3.7)

(10) Not required when not generating steam in the steam generators.

R TRM & LATER

⟨LATER⟩ (3.4B)

(11) The following shall be required until the end of Cycle 2 operation:

a. Gross radioiodine shall be determined at least three times per week during power operation.

LATER

3.4A

<LATER> (3.4B)

(2) A radiochemical analysis shall consist of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of Ē. A radiochemical analysis and calculation of Ē and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10 μCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes. LATER

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μCi/gm from the previous measured level.

<LATER> (3.4B & 3.7)

(4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity. LATER

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2. R TRM

<LATER> (3.4B)

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above. LATER

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prior to any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above.

<LATER> (3.1, 3.4B, 3.7)

(7) Not required when plant is in the cold shutdown condition or refueling shutdown condition. LATER & R TRM

(8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition. LA2 TRM

<LATER> (3.7)

(9) Required only when fuel is in the pool and prior to transferring fuel to the pool. LATER

<LATER> (3.7)

(10) Not required when not generating steam in the steam generators. LATER & R TRM

<LATER> (3.4B)

(11) The following shall be required until the end of Cycle 2 operation: LATER

a. Gross radioiodine shall be determined at least three times per week during power operation.

3.4.12

(2) A radiochemical analysis shall consist of the quantitative measurement the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of Ē. A radiochemical analysis and calculation of Ē and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10 µCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes. LA1 Bases L11 LA1 Bases

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 µCi/gm from the previous measured level. L11

& (LATER) (3.7) (4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity. LA1 Bases & LATER

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2. R TRM

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above. L11

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prior to any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above. & LA2 TRM

3.4.12 Appl. & LATER (3.1, 3.7) (7) ~~Not required when plant is in the cold shutdown condition or refueling shutdown condition.~~ MODES 1 & 2 & 3 w/ RCS T ≥ 500 F. L7 & LATER & R TRM

(LATER) 3.4A (8) O_2 analysis is not required when plant is in the cold shutdown conditic or refueling shutdown condition. LATER

(LATER) (3.7) (9) Required only when fuel is in the pool and prior to transferring fuel t the pool. LATER

(LATER) (3.7) (10) Not required when not generating steam in the steam generators. LATER & R TRM

(11) The following shall be required until the end of Cycle 2 operation: A6

a. Gross radioiodine shall be determined at least three times per weel during power operation.

3.7.4
3.7.13

<LATER> (3.4B)

(2) A radiochemical analysis shall consist of the quantitative measurement the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activitie: for the individual isotopes shall be used in the determination of $\bar{E}$. radiochemical analysis and calculation of $\bar{E}$ and iodine isotopic activi shall be performed if the measured gross activity changes by more than µCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall as given in "Table of Isotopes" (1967) and beta energy per disintegrat shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes. — LATER

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 µCi/gm from the previous measured level.

SR 3.7.4.1 & <LATER> (3.4B)

(4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity. — & LATER

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2. — R TRM

<LATER> (3.4B)

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less th: 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above. — LATER

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prio to any reactor criticality and analyzed for radioactive iodines of I-1: through I-135 and gross radioactivity as well as the coolant sample an analyses required by above.

3.7.4 Appl. & <LATER> (3.1 & 3.4B)

(7) Not required when plant is in the cold shutdown condition or refueling shutdown condition. — & R TRM & LATER

<LATER> (3.4A)

(8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition. — LATER

3.7.13 Appl.

(9) Required only when fuel is in the pool ~~and prior to transferring fuel to the pool.~~ — LA3 TRM

3.7.4 Appl.

(10) Not required when not generating steam in the steam generators. — A5 & R TRM

<LATER> (3.4B)

(11) The following shall be required until the end of Cycle 2 operation: — LATER

a. Gross radioiodine shall be determined at least three times per wee: during power operation.

Items on this page also addressed in the following packages: 3.1, 3.4A, 3.4B, 3.7

SR

(2) A radiochemical analysis shall consist of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of $\bar{E}$. A radiochemical analysis and calculation of $\bar{E}$ and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10 μCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes.

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μCi/gm from the previous measured level.

(4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity.

(5) ~~In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2.~~ R TRM

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above.

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prior to any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above.

(7) ~~Not required when plant is in the cold shutdown condition or refueling shutdown condition.~~ R TRM

(8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition.

(9) Required only when fuel is in the pool and prior to transferring fuel to the pool.

(10) ~~Not required when not generating steam in the steam generators.~~ R TRM

(11) The following shall be required until the end of Cycle 2 operation:

a. Gross radioiodine shall be determined at least three times per week during power operation.

3.4.12

b. ~~If the steady state gross radioiodine concentration increases by a factor of ten or more, the NRC shall be promptly notified with a written followup per Specification 6.12.3.1.~~

A6

5.0

4.2 REACTOR COOLANT SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the reactor coolant system pressure boundary.

A1

Objective

To assure the continued integrity of the reactor coolant system pressure boundary.

Specification

4.2.1 Prior to initial unit operation, an ultrasonic test survey shall be made of reactor coolant system pressure boundary welds as required to establish preoperational integrity and baseline data for future inspections.

A6

4.2.2 Post-operational inspections of components shall be made in accordance with the methods and intervals indicated in Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC.

A5

5.0

A5

~~4.2.3 The structural integrity of the reactor coolant system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a result of the tests outlined in Table IS-261 of Section XI of the code, that defects have developed or grown, shall be investigated.~~

LA6 SAR

~~4.2.4 To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support, shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.~~

LA6 QAPM

~~4.2.5 Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.~~

5.5.7

4.2.6 Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

A3

<INSERT 77>

A2

~~**Bases**~~

~~The surveillance program has been developed to comply with the applicable edition of Section XI and addenda of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, as required by 10CFR50.55a, to the extent practicable within limitations of design, geometry and materials of construction.~~

<CTS INSERT 77>

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program inspection frequencies.

4.3 TESTING FOLLOWING OPENING OF SYSTEM

<u>Applicability</u>

Applies to test requirements for Reactor Coolant System integrity.

<u>Objective</u>

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

<u>Specification</u>

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

⟨LATER⟩ (5.0) — LATER

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2155 psig, prior to the reactor being made critical, in accordance with the ASME Boiler and Pressure Vessel Codes, Section XI; IWA-5000.

LA1 Bases

4.3.3 The limitations of Specification 3.1.2 shall apply.

⟨LATER⟩ (5.0) — LATER

<u>Bases</u>

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes, such as B 31.7, and ASME Boiler and Pressure Vessel Code, Section XI.

For normal opening, the integrity of the Reactor Coolant System in terms of strength, is unchanged. The ASME Boiler and Pressure Vessel Code, Section XI; IWA-5000 requires a system leak test at nominal operating pressure (2155 psig) following system opening. At the end of refueling outages, this test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components.

<u>REFERENCES</u>

(1) FSAR, Section 4
(2) ASME Boiler and Pressure Vessel Code, Section XI

A2

5.0

(A5)

4.3 TESTING FOLLOWING OPENING OF SYSTEM

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

<LATER> (2.0)

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2155 psig, prior to the reactor being made critical, in accordance with the ASME Boiler and Pressure Vessel Code, Section XI; IWA-5000.

LATER

4.3.3 The limitations of Specification 3.1.2 shall apply.

A5

Bases

A2

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes, such as B 31.7, and ASME Boiler and Pressure Vessel Code, Section XI.

For normal opening, the integrity of the Reactor Coolant System in terms of strength, is unchanged. The ASME Boiler and Pressure Vessel Code, Section XI; IWA-5000 requires a system leak test at nominal operating pressure (2155 psig) following system opening. At the end of refueling outages, this test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components.

REFERENCES

(1) FSAR, Section 4
(2) ASME Boiler and Pressure Vessel Code, Section XI

3.6.1
3.6.2
3.6.3

3.6 **4.4 REACTOR BUILDING**

4.4.1 Reactor Building Leakage Tests

~~Applicability~~

~~Applies to the reactor building.~~

~~Objective~~

~~To verify that leakage from the reactor building is maintained within allowable limits.~~ A1

Specification

SR 3.6.1.1 4.4.1.1 Integrated leakage rate tests shall be conducted and visual inspections performed in accordance with the Reactor Building Leakage Rate Testing Program.

~~4.4.1.1.1 Deleted~~
~~4.4.1.1.2 Deleted~~
~~4.4.1.1.3 Deleted~~ A1

SR 3.6.1.1 4.4.1.1.4 Integrated leakage rate testing frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

~~4.4.1.1.5 Deleted~~
~~4.4.1.1.6 Deleted~~
~~4.4.1.1.7 Deleted~~ A1

SR 3.6.1.1
SR 3.6.2.1 4.4.1.2 Local leakage rate tests shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program.

~~4.4.1.2.1 Deleted~~
~~4.4.1.2.2 Deleted~~
~~4.4.1.2.3 Deleted~~
~~4.4.1.2.4 Deleted~~ A1

SR 3.6.1.1
SR 3.6.2.1 4.4.1.2.5 Local leakage rate testing frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program. A1

~~4.4.1.3 Deleted~~ LA3 IST

SR 3.6.3.4 4.4.1.4 Isolation Valve Functional Tests L18

~~Every three months,~~ ~~remotely operated~~ Automatic reactor building isolation valves shall be stroked, and timed, to the position required to fulfill their safety function ~~unless such operation is not practical during plant operation. The latter valves shall be tested once every 18 months.~~ A17 LA3 IST

~~4.4.1.5 Deleted~~ A1

<Add SR 3.6.2.1 Notes> M11

<Add SR 3.6.2.2> M12

3.6.1

A2

Bases (1)

The reactor building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 285°F.

The peak calculated reactor building pressure for the design basis loss of coolant accident, P_a, is 54 psig. The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of containment air weight per day at P_a.

The reactor building will be periodically leakage tested in accordance with the Reactor Building Leakage Rate Testing Program. These periodic testing requirements verify the reactor building leakage rate does not exceed the assumptions used in the safety analysis. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$.

REFERENCE

(1) FSAR, Sections 5 and 13.

<Add SR 3.5.2.1> M12

<Add SR 3.5.2.5> M13

4.5 EMERGENCY CORE COOLING SYSTEM AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 Emergency Core Cooling Systems

Applicability

Applies to periodic testing requirement for emergency core cooling systems. A1

Objective

To verify that the emergency core cooling systems are operable.

Specification

4.5.1.1 System Tests

SR 3.5.2.3
SR 3.5.2.4

4.5.1.1.1 High Pressure Injection System

(a) Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the high pressure injection system for emergency core cooling operation. LA3 SAR

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel. LA3 SAR

SR 3.5.2.3
SR 3.5.2.4
& <Later> (3.7)

4.5.1.1.2 Low Pressure Injection System

(a) Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. The test shall be performed in accordance with the procedure summarized below: & LATER LA3 SAR

(1) A test signal will be applied to demonstrate actuation of the low pressure injection system for emergency core cooling operation.

<LATER> (3.7)

(2) Verification of the engineered safeguard function of the service water system which supplies cooling water to the decay heat removal coolers shall be made to demonstrate operability of the coolers. LATER

& <LATER> (3.7)

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed, and all valves shall have completed their travel. LA3 SAR & LATER

4.5 EMERGENCY CORE COOLING SYSTEM AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

<LATER> (3.5)

~~4.5.1 Emergency Core Cooling Systems~~

~~Applicability~~

~~Applies to periodic testing requirement for emergency core cooling systems.~~

~~Objective~~

~~To verify that the emergency core cooling systems are operable.~~

~~Specification~~

~~4.5.1.1 System Tests~~

~~4.5.1.1.1 High Pressure Injection System~~

~~(a) Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the high pressure injection system for emergency core cooling operation.~~

~~(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel.~~

~~4.5.1.1.2 Low Pressure Injection System~~

LATER

SR 3.7.7.2 & <LATER> (3.5)

(a) Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. The test shall be performed in accordance with the procedure summarized below:

~~(1) A test signal will be applied to demonstrate actuation of the low pressure injection system for emergency core cooling operation.~~

SR 3.7.7.2

(2) Verification of the engineered safeguard function of the service water system which supplies cooling water to the decay heat removal coolers shall be made to demonstrate operability of the coolers. A1

& <LATER> (3.5)

~~(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed, and all valves shall have completed their travel.~~ LA3 TRM & LATER

3.5.2

4.5.1.1.3 Core Flooding System

(a) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. During this test, verification shall be made that the check valves in the core flooding tank discharge lines operate properly.

(b) The test will be considered satisfactory if control board indication of core flood tank level verifies that all check valves have opened.

LA3 SAR

4.5.1.2 Component Tests A1

4.5.1.2.1 Pumps

SR 3.5.2.2

Approximately quarterly the high pressure and low pressure injection pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ± 10% of the initial level of performance as determined using test flow paths.

LA3 SAR

4.5.1.2.2 Valves - Power Operated

<LATER> (3.7)

(a) At intervals not to exceed three months, each engineered safety feature valve in the emergency core cooling systems and each engineered safety feature valve associated with emergency core cooling in the service water system which are designed to open in the event of a LOCA shall be tested to verify operability.

LA3 SAR

LATER

¢<LATER> (3.7)

(b) The acceptable performance of each power operated valve will be that motion is indicated upon actuation by appropriate signals.

LA3 SAR ¢LATER

Bases

The emergency core cooling systems are the principle reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The high pressure injection system under normal operating conditions has one pump operating. At least once per month, operation will be rotated to another high pressure injection pump. This will help verify that the high pressure injection pumps are operable.

The requirements of the service water system for cooling water are more severe during normal operation than under accident conditions. Rotation of the pump in operation on a monthly basis will verify that two pumps are operable.

A2

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the borated water storage tank recirc line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

<LATER> (3.5)

4.5.1.1.3 Core Flooding System

(a) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. During this test, verification shall be made that the check valves in the core flooding tank discharge lines operate properly.

(b) The test will be considered satisfactory if control board indication of core flood tank level verifies that all check valves have opened.

LATER

4.5.1.2 Component Tests

4.5.1.2.1 Pumps

Approximately quarterly, the high pressure and low pressure injection pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ± 10% of the initial level of performance as determined using test flow paths.

4.5.1.2.2 Valves - Power Operated

(a) At intervals not to exceed three months, each engineered safety feature valve in the emergency core cooling systems and each engineered safety feature valve associated with emergency core cooling in the service water system which are designed to open in the event of a LOCA shall be tested to verify operability.

LA2 IST & LATER

& <LATER> (3.5)

(b) The acceptable performance of each power operated valve will be that motion is indicated upon actuation by appropriate signals.

Bases

The emergency core cooling systems are the principle reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The high pressure injection system under normal operating conditions has one pump operating. At least once per month, operation will be rotated to another high pressure injection pump. This will help verify that the high pressure injection pumps are operable.

A2

The requirements of the service water system for cooling water are more severe during normal operation than under accident conditions. Rotation of the pump in operation on a monthly basis will verify that two pumps are operable.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the borated water storage tank recirc line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

With the reactor shutdown, the check valves in each core flooding line are checked for operability by reducing the reactor coolant system pressure until the indicated level in the core flood tanks verify that the check valves have opened.

A2

<u>REFERENCE</u>

FSAR Section 6

3.6.5

4.5.2 Reactor Building Cooling Systems

~~Applicability~~

~~Applies to testing of the reactor building emergency cooling systems.~~ A1

~~Objective~~

~~To verify that the reactor building emergency cooling systems are operable.~~

Specification

4.5.2.1 System Tests

4.5.2.1.1 Reactor Building Spray System

SR 3.6.5.5
SR 3.6.5.6

(a) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. A test signal will be applied to demonstrate actuation of the reactor building spray system (except for reactor building inlet valves to prevent water entering nozzles).

SR 3.6.5.8

(b) ~~Station compressed air or smoke will be introduced into the spray headers to~~ verify the availability of the headers and spray nozzles at least every ~~five~~ 10 years. LA1 Bases L10

(c) ~~The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly.~~ LA3 IST

4.5.2.1.2 Reactor Building Cooling System 31 L11

SR 3.6.5.3

(a) At least once per ~~14~~ days, each reactor building emergency cooling train shall be tested to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

SR 3.6.5.3

(1) Verifying a service water flow rate of ≥ 1200 gpm to each train of the reactor building emergency cooling. A12

(2) ~~Addition of a biocide to the service water during the surveillance in 4.5.2.1.2.a.1 above, whenever service water temperature is between 60F and 80F.~~ LA1 Bases

SR 3.6.5.2

(b) At least once per 31 days, each reactor building emergency cooling train shall be tested to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) Starting (unless already operating) each operational cooling fan from the control room.

A12

~~Surveillance Requirement 4.5.2.1.2(a)(1) will not be performed on the green train of the reactor building emergency cooling system until cooling fan VSF-1D is repaired and the green train is returned to normal configuration. This note will remain in effect until July 14, 1995.~~

(2) Verifying that each operational cooling fan operates for at least 15 minutes.

SR 3.6.5.7 & LATER (3.7) (c) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

SR 3.6.5.7 (1) A test signal will be applied to actuate the reactor building emergency cooling operation.

<Later> (3.7) (2) ~~Verification of the engineered safety features function of the service water system which supplies the reactor building emergency coolers shall be made to demonstrate operability of the coolers.~~ Later

SR 3.6.5.7 & <Later> (3.7) (3) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly. LA1 Bases & Later

4.5.2.2 Component Tests

4.5.2.2.1 Pumps

SR 3.6.5.4 ~~At intervals not to exceed 3 months the reactor building spray pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ±10% of a point on the pump head curve.~~ LA3 IST

4.5.2.2.2 Valves

~~At intervals not to exceed three months each engineered safety features valve in the reactor building spray and reactor building emergency cooling system and each engineered safety features valve associated with reactor building emergency cooling in the service water system shall be tested to verify that it is operable.~~ LA3 IST <LATER> (3.7) LATER & <LATER> (3.7) LA3 IST & LATER

<Add SR 3.6.5.1> M15

~~Bases~~

~~The reactor building emergency cooling system and reactor building spray system are redundant to each other in providing post-accident cooling of the reactor building atmosphere to prevent the building pressure from exceeding the design pressure. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the reactor building emergency cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the reactor building spray system have been maintained consistent with that assigned other inoperable engineered safeguard equipment since the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere.~~ A2

<LATER> (3.6) (2) Verifying that each operational cooling fan operates for at least 15 minutes. LATER

SR 3.7.7.2 & <LATER> (3.6) (c) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to actuate the reactor building emergency cooling operation.

SR 3.7.7.2 (2) Verification of the engineered safety features function of the service water system which supplies the reactor building emergency coolers shall be made to demonstrate operability of the coolers. A1

SR 3.7.7.2 & <LATER> (3.6) (3) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly. LA3 TRM & LATER

<LATER> (3.6) 4.5.2.2 Component Tests LATER

4.5.2.2.1 Pumps

At intervals not to exceed 3 months the reactor building spray pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ±10% of a point on the pump head curve.

4.5.2.2.2 Valves

At intervals not to exceed three months each engineered safety features valve in the reactor building spray and reactor building emergency cooling system and each engineered safety features valve associated with reactor building emergency cooling in the service water system shall be tested to verify that it is operable. LA2 IST

Bases A2

The reactor building emergency cooling system and reactor building spray system are redundant to each other in providing post-accident cooling of the reactor building atmosphere to prevent the building pressure from exceeding the design pressure. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the reactor building emergency cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the reactor building spray system have been maintained consistent with that assigned other inoperable engineered safeguard equipment since the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere.

3.6.5

Addition of ~~a biocide to service water is performed during reactor building emergency cooler surveillance to prevent buildup of Asian clams in the coolers when service water is pumped through the cooling coils. This is performed when service water temperature is between 60F and 80F since in this water temperature range Asian clams can spawn and produce larva which could pass through service water system strainers.~~

~~The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.~~

~~With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or smoke can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.~~

~~The equipment, piping, valves, and instrumentation of the reactor building emergency cooling system are arranged so that they can be visually inspected. The cooling fans and coils and associated piping are located outside the secondary concrete shield. Personnel can enter the reactor building during power operations to inspect and maintain this equipment. The service water piping and valves outside the reactor building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.~~

~~Two service water pumps are normally operating. At least once per month operation of one pump is shifted to the third pump, so testing will be unnecessary.~~

~~As the reactor building fans are normally operating, starting for testing is unnecessary for those verified to be operating.~~

A2

~~**Reference**~~

~~FSAR, Section 6~~

3.7.7

A2

Addition of a biocide to service water is performed during reactor building emergency cooler surveillance to prevent buildup of Asian clams in the coolers when service water is pumped through the cooling coils. This is performed when service water temperature is between 60F and 80F since in this water temperature range Asian clams can spawn and produce larva which could pass through service water system strainers.

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or smoke can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves, and instrumentation of the reactor building emergency cooling system are arranged so that they can be visually inspected. The cooling fans and coils and associated piping are located outside the secondary concrete shield. Personnel can enter the reactor building during power operations to inspect and maintain this equipment. The service water piping and valves outside the reactor building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.

Two service water pumps are normally operating. At least once per month operation of one pump is shifted to the third pump, so testing will be unnecessary.

As the reactor building fans are normally operating, starting for testing is unnecessary for those verified to be operating.

Reference

FSAR, Section 6

3.8.1
3.8.2

4.6 AUXILIARY ELECTRICAL SYSTEM TESTS

~~Applicability~~ (A1)

~~Applies to the periodic testing and surveillance requirements of the auxiliary electrical system to ensure it will respond promptly and properly when required.~~

~~Specification~~

4.6.1 Diesel Generators (LA3 TRM)

SR 3.8.1.2 SR 3.8.1.3
1. Each diesel generator shall be ~~manually~~ started each month and demonstrated to be ready for loading within 15 seconds. ~~The signal initiating the start of the diesel shall be varied from one test to another (start with handswitch at control room panel and at diesel local control panel) to verify all starting circuits are operable.~~ (LA1 Bases) The generator shall be synchronized from the control room and loaded to full rated load and allowed to run ~~until diesel generator operating temperatures have stabilized.~~ ≥ 1 hour (A9) (M15)

SR 3.8.1.7 SR 3.8.1.8
2. A test shall be conducted once every 18 months to demonstrate the ability of the diesel generators to perform as designed by: (M12)

SR 3.8.1.7
a. simulating a loss of off-site power, (M13)

SR 3.8.1.8
b. simulating of loss of off-site power in conjunction with an ESF signal,

~~c. simulating interruption of off-site power and subsequent reconnection of the on-site power source to their respective busses, and~~ (L8)

SR 3.8.1.7 SR 3.8.1.8
d. operating the diesel generator for ~~≥1 hour after operating temperatures have stabilized.~~ 5 minutes (L7)

~~3. Each diesel generator shall be given an inspection once every 18 months following the manufacturer's recommendations for this class of standby service.~~ (LA1 Bases) ~~(A one-time extension of this interval is allowed so that these may be performed during the 1R9 refueling outage, and completed no later than December 1, 1990.)~~ (A4)

& (LATER) (5.0)
4. ~~During the~~ monthly ~~diesel generator test specified in paragraph 1 above,~~ the following shall be performed: (L11 & LATER)

SR 3.8.2.3
a. The diesel generator starting air compressors shall be ~~checked for operation and their ability to recharge the~~ air receivers. (L9)

SR 3.8.1.5
b. The diesel oil transfer pumps shall be checked for operability and their ability to transfer oil to the day tank.

SR 3.8.1.4
c. The day tank fuel level shall be verified.

SR 3.8.2.1
d. The emergency storage tank fuel level shall be verified.

5.5.13

4.6 AUXILIARY ELECTRICAL SYSTEM TESTS

Applicability

Applies to the periodic testing and surveillance requirements of the auxiliary electrical system to ensure it will respond promptly and properly when required.

Specification

4.6.1 Diesel Generators

(LATER) (3.8)

1. Each diesel generator shall be manually started each month and demonstrated to be ready for loading within 15 seconds. The signal initiating the start of the diesel shall be varied from one test to another (start with handswitch at control room panel and at diesel local control panel) to verify all starting circuits are operable. The generator shall be synchronized from the control room and loaded to full rated load and allowed to run until diesel generator operating temperatures have stabilized. LATER

2. A test shall be conducted once every 18 months to demonstrate the ability of the diesel generators to perform as designed by:

   a. simulating a loss of off-site power,

   b. simulating of loss of off-site power in conjunction with an ESF signal,

   c. simulating interruption of off-site power and subsequent reconnection of the on-site power source to their respective busses, and

   d. operating the diesel generator for ≥1 hour after operating temperatures have stabilized.

3. Each diesel generator shall be given an inspection once every 18 months following the manufacturer's recommendations for this class of standby service. (A one-time extension of this interval is allowed so that these may be performed during the 1R9 refueling outage, and completed no later than December 1, 1990.)

5.5.13 & LATER (3.8)

4. During the monthly diesel generator test specified in paragraph 1 above, the following shall be performed: L8 & LATER

(LATER) (3.8)

   a. The diesel generator starting air compressors shall be checked for operation and their ability to recharge the air receivers.

   b. The diesel oil transfer pumps shall be checked for operability and their ability to transfer oil to the day tank. LATER

   c. The day tank fuel level shall be verified.

   d. The emergency storage tank fuel level shall be verified.

3.8.2
3.8.3
3.8.4

<ADD LCO 3.8.2 ACTION B,C & SR 3.8.2.2 for New fuel oil> M11

SR 3.8.2.2 & <LATER>(5.0)

e. Diesel fuel from the emergency storage tank shall be sampled and found to be within acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water, and sediment. LATER

SR 3.8.2.3 5. Once every 31 days the pressure in the required starting air receiver tanks shall be verified to be ≥ 175 psig.

~~Once every 18 months, the capacity of each diesel oil transfer pump shall be verified to be at least 10 gpm.~~ LA4

4.6.2 DC Sources and Battery Cell Parameters

SR 3.8.3.1 1. Verify battery terminal voltage is ≥ 124.7 V on float charge once each 7 days.

SR 3.8.3.2 2. Verify battery capacity is adequate to supply, and maintain in operable status, the required emergency loads for the design duty cycle when subjected to either a battery service test or a modified performance discharge test once every 18 months.

SR 3.8.3.3 3. Verify battery capacity is ≥ 80% of the manufacturers rating when subjected to a performance discharge test or a modified performance discharge test once every 60 months, once every 24 months when battery has reached 85% of the service life with capacity ≥ 100% of the manufacturers rating and showing no degradation, and once every 12 months when battery shows degradation or has reached 85% of the service life and capacity is < 100% of the manufacturer's rating.

~~4. Any battery charger which has not been loaded while connected to its 125V d-c distribution system for at least 30 minutes during every quarter shall be tested and loaded while connected to its bus for 30 minutes.~~ LA5

SR 3.8.4.1 5. Verify battery pilot cell parameters meet Table 4.6-1 Category A limits once per 7 days.

SR 3.8.4.4 6. Verify average electrolyte temperature of representative cells is ≥ 60°F once per 92 days.

SR 3.8.4.3 7. Verify battery cell parameters meet Table 4.6-1 Category B limits once per 92 days and once within 24 hours after a battery discharge to < 110 V and once within 24 hours after a battery overcharge to > 145 V.

SR 3.8.4.2 8. Verify electrolyte temperature of pilot cell is ≥ 60°F once per 31 days.

~~4.6.3 Emergency Lighting~~

~~The correct functioning of the emergency lighting system shall be verified once every 18 months.~~ R TRM

<INSERT CTS100aA>

for ITS 3.8.1 AC Sources - Operating

Add SR 3.8.1.1 M9

Add SR 3.8.1.2 NOTE A7

Add SR 3.8.1.3 NOTE 1 A7

Add SR 3.8.1.3 NOTE 2 A7

Add SR 3.8.1.3 NOTE 3 A7

Add SR 3.8.1.7 NOTE A7

Add SR 3.8.1.8 NOTE A7

<INSERT CTS100aB>

for ITS 3.8.6 Distribution Systems - Operating

Add SR 3.8.6.1 M14

5.5.13

5.5.13 <ADD: Diesel Fuel Oil Testing Program description> — LA3

<ADD: SR 3.0.2 & SR 3.0.3 Applicability statement> — A3

<ADD: New fuel oil testing> — M9

e. Diesel fuel from the emergency storage tank shall be sampled and found to be within acceptable limits ~~specified in Table 1 of ASTM D975-68 when checked for viscosity, water, and sediment.~~ — L6

~~5. Once every 31 days the pressure in the required starting air receiver tanks shall be verified to be ≥ 175 psig.~~

~~Once every 18 months, the capacity of each diesel oil transfer pump shall be verified to be at least 10 gpm.~~

<LATER> (3.8)

~~4.6.2 DC Sources and Battery Cell Parameters~~ — LATER

~~1. Verify battery terminal voltage is ≥ 124.7 V on float charge once each 7 days.~~

~~2. Verify battery capacity is adequate to supply, and maintain in operable status, the required emergency loads for the design duty cycle when subjected to either a battery service test or a modified performance discharge test once every 18 months.~~

~~3. Verify battery capacity is ≥ 80% of the manufacturers rating when subjected to a performance discharge test or a modified performance discharge test once every 60 months, once every 24 months when battery has reached 85% of the service life with capacity ≥ 100% of the manufacturers rating and showing no degradation, and once every 12 months when battery shows degradation or has reached 85% of the service life and capacity is < 100% of the manufacturer's rating.~~

~~4. Any battery charger which has not been loaded while connected to its 125V d-c distribution system for at least 30 minutes during every quarter shall be tested and loaded while connected to its bus for 30 minutes.~~

~~5. Verify battery pilot cell parameters meet Table 4.6-1 Category A limits once per 7 days.~~

~~6. Verify average electrolyte temperature of representative cells is ≥ 60°F once per 92 days.~~

~~7. Verify battery cell parameters meet Table 4.6-1 Category B limits once per 92 days and once within 24 hours after a battery discharge to < 110 V and once within 24 hours after a battery overcharge to > 145 V.~~

~~8. Verify electrolyte temperature of pilot cell is ≥ 60°F once per 31 days.~~

~~4.6.3 Emergency Lighting~~

~~The correct functioning of the emergency lighting system shall be verified once every 18 months.~~ — R TRM

e. Diesel fuel from the emergency storage tank shall be sampled and found to be within acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water, and sediment.

5. Once every 31 days the pressure in the required starting air receiver tanks shall be verified to be ≥ 175 psig.

   Once every 18 months, the capacity of each diesel oil transfer pump shall be verified to be at least 10 gpm.

4.6.2 DC Sources and Battery Cell Parameters

1. Verify battery terminal voltage is ≥ 124.7 V on float charge once each 7 days.

2. Verify battery capacity is adequate to supply, and maintain in operable status, the required emergency loads for the design duty cycle when subjected to either a battery service test or a modified performance discharge test once every 18 months.

3. Verify battery capacity is ≥ 80% of the manufacturers rating when subjected to a performance discharge test or a modified performance discharge test once every 60 months, once every 24 months when battery has reached 85% of the service life with capacity ≥ 100% of the manufacturers rating and showing no degradation, and once every 12 months when battery shows degradation or has reached 85% of the service life and capacity is < 100% of the manufacturer's rating.

4. Any battery charger which has not been loaded while connected to its 125V d-c distribution system for at least 30 minutes during every quarter shall be tested and loaded while connected to its bus for 30 minutes.

5. Verify battery pilot cell parameters meet Table 4.6-1 Category A limits once per 7 days.

6. Verify average electrolyte temperature of representative cells is ≥ 60°F once per 92 days.

7. Verify battery cell parameters meet Table 4.6-1 Category B limits once per 92 days and once within 24 hours after a battery discharge to < 110 V and once within 24 hours after a battery overcharge to > 145 V.

8. Verify electrolyte temperature of pilot cell is ≥ 60°F once per 31 days.

4.6.3 Emergency Lighting

The correct functioning of the emergency lighting system shall be verified once every 18 months.

TRM

TABLE 3.8.4-1

Table ~~4.6-1~~ 3.8.4-1 (page 1 of 1)
Battery Cell Surveillance Requirements

A1

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ 1/4 inch above maximum level indication mark[a]	> Minimum level indication mark, and ≤ 1/4 inch above maximum level indication mark[a]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[b][c]	≥ 1.195	≥ 1.190 **AND** Average of all connected cells > 1.195	Not more than 0.020 below average connected cells **AND** Average of all connected cells ≥ 1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

3.8.1
3.8.3
3.8.4

Bases

The emergency power system provides power requirements for the engineered safety features in the event of a DBA. Each of the two diesel generators is capable of supplying minimum required engineered safety features from independent buses. This redundancy is a factor in establishing testing intervals. The monthly tests specified above will demonstrate operability and load capacity of the diesel generator. The fuel supply and diesel starter motor air pressure are continuously monitored and alarmed for abnormal conditions. Starting on complete loss of off-site power will be verified by simulated loss-of-power tests once every 18 months.

A2

The SR 4.6.2.1 verification of battery terminal voltage while on float charge helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the battery charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery (2.15 V per cell average) and are consistent with the battery vendor allowable minimum volts per cell. The inability to meet this requirement constitutes an inoperable battery.

The SR 4.6.2.2 battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements. A modified performance discharge test may be performed in lieu of a service test. The inability to meet this requirement constitutes an inoperable battery.

The modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the battery. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified performance discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test and the test discharge rate must envelope the duty cycle of the service test if the modified performance discharge test is performed in lieu of a service test.

The SR 4.6.2.3 battery performance discharge test is a test of constant current capacity of a battery after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage. The inability to meet this requirement constitutes an inoperable battery.

Either the battery performance discharge test or the modified performance discharge test, described above, is acceptable for satisfying SR 4.6.2.3; however, only the modified performance discharge test may be used to satisfy SR 4.6.2.3 while satisfying the requirements of SR 4.6.2.2 at the same time.

3.8.3
3.8.4

The acceptance criteria for this surveillance are consistent with IEEE-450. This reference recommends that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The frequency for this test is normally 60 months. If the battery shows signs of degradation, or if the battery has reached 85% of its service life and capacity is < 100% of the manufacturer's rating, the frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its service life, the frequency is only reduced to 24 months for batteries that retain ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450, when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer's rating.

A2

SR 4.6.2.4 requires that each required battery charger be capable of supplying the connected loads while maintaining the battery fully charged. This is based on the assumption that the batteries are fully charged at the beginning of a design basis accident, and on the safety function of providing adequate power for the design basis accident loads.

SR 4.6.2.5 verifies that the Table 4.6-1 Category A battery cell parameters are consistent with vendor recommendations and IEEE-450, which recommend regular battery inspections (at least once per month) including voltage, specific gravity, and electrolyte level of pilot cells.

The SR 4.6.2.6 verification that the average temperature of representative cells is ≥ 60°F is consistent with a recommendation of IEEE-450, which states that the temperature of electrolytes in representative cells (~10% of all connected cells) should be determined on a quarterly basis. Lower than normal temperatures act to inhibit or reduce battery capacity. This surveillance ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

SR 4.6.2.7 verifies that the Table 4.6-1 Category B battery cell parameters are consistent with vendor recommendations and IEEE-450, which recommend regular battery inspections (at least once per quarter) including voltage, specific gravity, and electrolyte level of each connected cell. In addition, within 24 hours after a battery discharge to < 110 V or a battery overcharge to > 145 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to ≤ 110 V, do not constitute a battery discharge provided battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450, which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

The SR 4.6.2.8 verification that the temperature of the pilot cell is ≥ 60°F is consistent with a recommendation of IEEE-450, which states that the temperature of electrolytes in pilot cells should be determined on a monthly basis. Lower than normal temperatures act to inhibit or reduce battery capacity. This surveillance ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

Table 4.6-1 delineates the limits on electrolyte level, cell float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage and electrolyte specific gravity approximate the state of charge of the entire battery.

3.8.3
3.8.4

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450, with the extra 1/4 inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote (a) to Table 4.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the battery vendor allowable minimum cell voltage and on a recommendation of IEEE-450, which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.195. This value is characteristic of a charged cell with adequate capacity. According to IEEE-450, the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that is jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.190 with the average of all connected cells > 1.195. These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery.

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limit for float voltage is based on IEEE-450, which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limits of average specific gravity ≥ 1.190 is based on manufacturer recommendations. In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

A2

3.8.4

Footnotes (b) and (c) to Table 4.6-1 are ~~applicable to Category A, B, and C~~ specific gravity. Footnote (b) to Table 4.6-1 requires the above mentioned correction for electrolyte temperature. The value of 2 amps used in footnote (c) is the nominal value for float current established by the battery vendor as representing a fully charged battery with an allowance for overall battery condition. This current provides, in general, an indication of overall battery condition.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450. Footnote (c) to Table 4.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. Within 7 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

A2

R

TRM

The SR 4.6.3 testing of the emergency lighting is scheduled every 18 months and is subject to review and modification if experience demonstrates a more effective test schedule.

REFERENCE

FSAR, Section 8

SR

Items on this page are also addressed in package 3.8

Footnotes (b) and (c) to Table 4.6-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 4.6-1 requires the above mentioned correction for electrolyte temperature. The value of 2 amps used in footnote (c) is the nominal value for float current established by the battery vendor as representing a fully charged battery with an allowance for overall battery condition. This current provides, in general, an indication of overall battery condition.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450. Footnote (c) to Table 4.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. Within 7 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

The SR 4.6.3 testing of the emergency lighting is scheduled every 18 months and is subject to review and modification if experience demonstrates a more effective test schedule.

R

TRM

<u>REFERENCE</u>

FSAR, Section 8

3.1.4
3.1.6
3.1.7

<Add SR 3.1.4.3 Note> L2

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 Control Rod Drive System Functional Tests

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

A1

criticality L2

M8

M17

reactor vessel head removal

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow ~~or no flow~~ conditions following each ~~refueling outage~~ prior to ~~return to power~~. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the Axial Power Shaping Rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66 seconds at reactor coolant ~~full~~ flow conditions ~~or 1.20 seconds for no flow conditions. For the APSRs it shall be demonstrated that loss of power will not cause rod movement.~~ If the trip insertion time above is not met, the rod shall be declared inoperable.

SR 3.1.4.3

LA1 SAR

6.5%

A3

A11

3.1.4 LCO

3.1.6 LCO

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine (9) inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. ~~The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.~~

LA1 BASES

M18

A6

4.7.1.3 If a control rod cannot be exercised, ~~or if it cannot be located with absolute or relative position indications or in or out limit lights,~~ the rod shall be declared to be inoperable.

LA1 BASES

3.1.7 RA A.1

Bases

~~The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR, Section 14, whose calculations are based on a rod drop from fully withdrawn to 2/3 inserted. Since the most accurate position indication is obtained from the zone reference switch at the 3/4 inserted position, this position is used instead of the 2/3 inserted position for data gathering.~~

~~Each control rod drive mechanism shall be exercised by a movement approximately two (2) inches of travel every two (2) weeks. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.~~

~~A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod~~

A2

<Add 3.1.7 LCO> L11

<Add 3.1.7 Appl.> M13

<Add 3.1.7 Actions Note> A12

3.1.4

~~deviates from its group average position by more than nine (9) inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2.~~

~~**REFERENCE**~~

~~(1) FSAR, Section 14~~

A2

Items on this page also addressed in the following packages: NA

SR

R

SAR

4.7.2 Control Rod Program Verification (Group Vs Core Positions)

Applicability

Applies to surveillance of the control rod systems.

Objective

To verify that the designated control rod (by core position) is operating in its programmed functional position and group (rods 1 through 12, group 1-8).

Specification

4.7.2.1 Whenever the control rod drive patch panel is reconnected (after test, reprogramming, or maintenance), each control rod drive mechanism shall be selected from the control room and exercised by movement of sufficient travel to verify that the proper rod has responded as shown on the unit computer printout or on the input to the computer for that rod.

4.7.2.2 Whenever power or instrumentation cables to the control rod drive assemblies atop the reactor or at the bulkhead are disconnected or removed, an independent verification check of their reconnection shall be performed.

4.7.2.3 Any rod found to be improperly programmed shall be declared inoperable until properly programmed.

Bases

Each control rod has a relative and an absolute position indicator system. One set of outputs goes to the plant computer identified by a unique number associated with only one core position. The other set of outputs goes to a programmable bank of 68 edgewise meters in the control room. In the event that a patching error is made in the patch panel or connectors in the cables leading to the control rod drive assemblies or the control room meter bank is improperly transposed upon reconnection, these errors and transpositions will be discovered by a comparative check by (1) selecting a specific rod from one group (e.g., rod 1 in regulating group 6), (2) noting the program-approved core position for this rod of the group, (3) exercising the selected rod, and (4) noting that a) the computer prints out both absolute and relative position response for the approved core position, and b) the proper meter in the control room display bank indicates both absolute and relative meter positions. This type of comparative check will not assure detection of improperly connected cables inside the reactor building. For these, (Specification 4.7.2.2) it will be necessary for a responsible person, other than the one doing the work, to verify by appropriate means that each cable has been matched to the proper control rod drive assembly.

3.7.5

4.8 EMERGENCY FEEDWATER PUMP TESTING

~~Applicability~~

~~Applies to the periodic testing of the turbine and electric motor driven emergency feedwater pumps.~~

~~Objective~~

~~To verify that the emergency feedwater pump and associated valves are operable.~~

A1

Specification

4.8.1 Each EFW train shall be demonstrated operable:

L9

a) ~~By verifying on a STAGGERED TEST BASIS:~~

M26

SR 3.7.5.2 & Note

1. ~~at least once per 31 days~~ ~~or within 24 hours after reaching the Hot Shutdown condition following a plant heatup and prior to criticality,~~ that the turbine-driven pump ~~starts,~~ operates ~~for a minimum of 5 minutes and develops a discharge pressure of ≥ 1200 psig at a flow of ≥ 500 gpm through the test loop flow path.~~

L10

LA2 IST

SR 3.7.5.2

2. ~~at least once per 31 days~~ by verifying that the motor driven EFW pump ~~starts,~~ operates ~~for a minimum of 5 minutes and develops a discharge pressure of ≥ 1200 psi at a flow of ≥ 500 gpm through the test loop flow path.~~

L9

LA2 IST

SR 3.7.5.1

b) At least once per 31 days by verifying that each valve (manual, power operated or automatic) in each EFW flowpath that is not locked, sealed, or otherwise secured in position is its correct position.

A3

SR 3.7.5.5

c) Prior to <entering MODE 2> ~~relying upon any steam generator for heat removal~~ whenever the plant has been in ~~CSD or less~~ MODE 5, MODE 6, or defueled for > 30 days, verify proper alignment of each manual valve in each required EFW flow path, which if mispositioned may degrade EFW operation, from the 'Q' condensate storage tank to each steam generator.

L12

A1

d) ~~At least once per 92 days by cycling each motor-operated valve in each flowpath through at least one complete cycle.~~

LA2 IST

e) At least once per 18 months by functionally testing each EFW train and:

SR 3.7.5.3

1. Verifying that each automatic valve in each flowpath actuates automatically to its correct position on receipt of an actual or simulated actuation signal.

<Add SR 3.7.5.3, Note>

L10

<Add SR 3.7.5.4, Note>

L10

3.7.5

SR 3.7.5.3
SR. 3.7.5.4

2. Verifying that the automatic steam supply valves associated with the steam turbine driven EFW pump actuate to their correct positions upon receipt of an actual or simulated actuation signal. ~~This test is not required to be performed until 24 hours after reaching the Hot Shutdown condition.~~ M27

SR 3.7.5.4

3. Verifying that the motor-driven EFW pump starts automatically upon receipt of an actual or simulated actuation signal.

SR 3.7.5.6

4. Verifying that feedwater is delivered to each steam generator using the electric motor-driven EFW pump.

SR 3.7.5.3

5. Verifying that the EFW system can be operated manually by over-riding automatic signals to the EFW valves. LA1 — Bases

A2

~~Bases~~

~~The monthly testing frequency will be sufficient to verify that both emergency feedwater pumps are operable. Verification of correct operation will be made both from the control room instrumentation and direct visual observation of the pumps. The cycling of the emergency valves assures valve operability when called upon to function. Testing of the turbine dri[illegible] EFW pump is delayed until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perfor[illegible] the test at 280°F. Testing may occur at a lower steam generator pressure if operational experience shows that sufficient steam pressure to perform the [illegible] exists.~~

~~Surveillance Requirement 4.8.1.c ensures that the EFW system is properly al[illegible] by verifying the flow paths to each steam generator prior to relying upon a[illegible] steam generator for heat removal after more than 30 days in Cold Shutdown o[illegible] below. Operability of the EFW flow paths must be demonstrated before suffi[illegible] core heat is generated that would require the operation of the EFW system o[illegible] a subsequent shutdown. This requirement is reasonable, based on engineerin[illegible] judgment, in view of other administrative controls to ensure that the flow [illegible] are operable. To further ensure EFW system alignment, flow path operabilit[illegible] verified following extended outages to determine no misalignment of valves [illegible] occurred. This SR ensures that the flow path from the 'Q' CST to the steam generators is properly aligned.~~

~~The functional test, performed once every 18 months, will verify that the flow path to the steam generators is open and that water reaches the steam generators from the emergency feedwater system. The test is done during shutdown to avoid thermal cycle to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater.~~

~~The automatic actuation circuitry testing and calibration will be performed per Surveillance Specification 4.1, and will be sufficient to assure that this circuitry will perform its intended function when called upon.~~

3.1.2 — 4.9 REACTIVITY ~~ANOMALIES~~ BALANCE A1

~~Applicability~~

~~Applies to potential reactivity anomalies.~~

~~Objective~~ A1

~~To require the evaluation of reactivity anomalies of a specified magnitude~~ occurring during the operation of the unit. A1

3.1.2 Appl. — Specification MODES 1 and 2

SR 3.1.2.1 — Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. M1

3.1.2 LCO — If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation of this abnormal

3.1.2 RA A.1 — occurrance will be made to determine the cause of the discrepancy. within 7 days M2

~~Bases~~

~~To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration necessary to maintain adequate control characteristics must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10 percent of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1 percent Δk/k would be unexpected, and its occurrence would be thoroughly investigated and evaluated.~~ A2

~~The value of 1 percent Δk/k is considered a safe limit since a shutdown margin of at least 1 percent Δk/k with the most reactive rod in the fully withdrawn position is always maintained.~~

<Add 3.1.2 RA A.2> M15

<Add 3.1.2 RA B.1> M15

<Add SR 3.1.2.1 NOTE and Frequency with NOTE> M15

3.7.9
3.7.10

4.10 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEM SURVEILLANCE

Applicability — A1

Applies to the surveillance of the control room emergency ventilation and air conditioning systems.

Objective

To verify an acceptable level of efficiency and operability of the control room emergency ventilation and air conditioning systems.

Specification

4.10.1 Each train of control room emergency air conditioning shall be demonstrated Operable:

SR 3.7.10.1

a. At least once per 31 days ~~on a staggered test basis~~ by: — LA1 BASES

1. Starting each unit and
2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤84°F D.B.

SR 3.7.10.2

b. At least once per 18 months by verifying a system flow rate of 9900 cfm ±10%.

4.10.2 Each Control Room Emergency Ventilation System shall be demonstrated Operable:

SR 3.7.9.1

a. At least once per 31 days on a Staggered Test Basis by initiating, from the Control Room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes. — LA1 Bases

SR 3.7.9.2
+ <LATER>
(5.0)

b. At least once per 18 months or 1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or 2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by: — LATER

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ±10%.
2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

   a. ≤2.5% for 2 inch charcoal adsorber beds, or

   b. ≤0.5% for 4 inch charcoal adsorber beds.
3. Verifying a system flow rate of 2000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

5.5.11

4.10 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEM SURVEILLANCE

LATER

<LATER> (3.7)

Applicability

Applies to the surveillance of the control room emergency ventilation and air conditioning systems.

Objective

To verify an acceptable level of efficiency and operability of the control room emergency ventilation and air conditioning systems.

Specification

4.10.1 Each train of control room emergency air conditioning shall be demonstrated Operable:

a. At least once per 31 days on a staggered test basis by:

1. Starting each unit and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤84°F D.B.

b. At least once per 18 months by verifying a system flow rate of 9900 cfm ±10%.

4.10.2 Each Control Room Emergency Ventilation System shall be demonstrated Operable:

a. At least once per 31 days on a Staggered Test Basis by initiating, from the Control Room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

5.5.11 & <LATER> (3.7) b. At least once per 18 months or 1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or 2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by: A8 & LATER

5.5.11.a.2 5.5.11.b.2 1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and ~~uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,~~ and the system flow rate is 2000 cfm ±10%. A8

5.5.11.c.3 2. Verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of: LA3 VFTP

a. ≤2.5% for 2 inch charcoal adsorber beds, or

b. ≤0.5% for 4 inch charcoal adsorber beds.

5.5.11.a.2 5.5.11.b.2 5.5.11.d 3. Verifying a system flow rate of 2000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

SR 3.7.9.2
+<LATER>
(5.0)

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

1. ≤ 2.5% for 2 inch charcoal adsorber beds, or

2. ≤ 0.5% for 4 inch charcoal adsorber beds.

+LATER

SR 3.7.9.2
SR 3.7.9.3
+<LATER (5.0)>

d. At least once per 18 months by:

+LATER

SR 3.7.9.2
+<LATER (5.0)>

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches of water while operating at a flowrate of 2000 cfm ±10%.

SR 3.7.9.3

2. Verifying that on a Control Room high radiation test signal, the system automatically isolates the Control Room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

L16

LA1
Bases

SR 3.7.9.2

+ <LATER>
(5.0)

e. After each complete or partial replacement of the HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%.

+LATER

Bases

A2

The purpose of the control room emergency ventilation system is to limit the particulate and gaseous fission products to which the control area would be subjected during an accidental radioactive release in or near the Auxiliary Building. The system is designed with 100 percent capacity filter trains which consist of a prefilter, high efficiency particulate filters, charcoal adsorbers and a fan.

Since the emergency ventilation system is not normally operated, a periodic test is required to insure operability when needed. During this test the system will be inspected for such things as water, oil, or other foreign material; gasket deterioration, adhesive deterioration in the HEPA units; and unusual or excessive noise or vibration when the fan motor is running. Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 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.

5.5.11

5.5.11
5.5.11.c.3

c. ~~After every 720 hours of charcoal adsorber operation~~ by verifying ~~within 31 days after removal~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of: A8 LA3

<LATER> (3.7)

1. ≤ 2.5% for 2 inch charcoal adsorber beds, or

2. ≤ 0.5% for 4 inch charcoal adsorber beds.

5.5.11
5.5.11.d

d. ~~At least once per 18 months~~ by: A8

1. Verifying that the pressure ~~drop across the combined HEPA filters~~ other filters in the system and charcoal adsorber banks is < 6 inches of water while operating at a flowrate of 2000 cfm ±10%. M5

<LATER> (3.7)

~~2. Verifying that on a Control Room high radiation test signal, the system automatically isolates the Control Room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.~~ LATER

5.5.11
5.5.11.a.2

e. ~~After each complete or partial replacement of the HEPA filter bank~~ by verifying that the HEPA filter banks remove ≥99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%. A8

<LATER> (3.7)

5.5.11
5.5.11.b.2

f. ~~After each complete or partial replacement of a charcoal adsorber bank~~ by verifying that the charcoal adsorbers remove ≥99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%. A8

~~**Bases**~~

~~The purpose of the control room emergency ventilation system is to limit the particulate and gaseous fission products to which the control area would be subjected during an accidental radioactive release in or near the Auxiliary Building. The system is designed with 100 percent capacity filter trains which consist of a prefilter, high efficiency particulate filters, charcoal adsorbers and a fan.~~

~~Since the emergency ventilation system is not normally operated, a periodic test is required to insure operability when needed. During this test the system will be inspected for such things as water, oil, or other foreign material; gasket deterioration, adhesive deterioration in the HEPA units; and unusual or excessive noise or vibration when the fan motor is running. Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 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.~~ A2

<ADD: SR 3.0.2 & SR 3.0.3 applicability statement> A3

Bases (Continued)

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d of Regulatory Guide 1.52. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbent qualified according to Regulatory Guide 1.52. A2

The operability of the control room emergency air conditioning Systems ensure that the ambient air temperature does not exceed the allowable temperature for the equipment and instrumentation cooled by this system and the Control Room will remain habitable for Operations personnel during and following all credible accident conditions.

Operation of the systems for 15 minutes every month will demonstrate operability of the emergency ventilation and emergency air conditioning systems. All dampers and other mechanical and isolation systems will be shown to be operable.

If significant painting, fire or chemical release occurs such that the HEPA filter of charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

5.5.11

A.2

Bases (Continued)

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d of Regulatory Guide 1.52. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbent qualified according to Regulatory Guide 1.52.

The operability of the control room emergency air conditioning Systems ensure that the ambient air temperature does not exceed the allowable temperature for the equipment and instrumentation cooled by this system and the Control Room will remain habitable for Operations personnel during and following all credible accident conditions.

Operation of the systems for 15 minutes every month will demonstrate operability of the emergency ventilation and emergency air conditioning systems. All dampers and other mechanical and isolation systems will be shown to be operable.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

3.7.11

4.11 PENETRATION ROOM VENTILATION SYSTEM SURVEILLANCE

~~Applicability~~

~~Applies to the surveillance of the penetration room ventilation system.~~ A1

~~Objective~~

~~To verify an acceptable level of efficiency and operability of the penetration room ventilation system.~~

Specification

SR 3.7.11.2 & (LATER) (5.0)

~~4.11.1 At intervals not to exceed 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (± 10%).~~ LATER

~~4.11.2 Initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers.~~

SR 3.7.11.3 4.11.3 At intervals not to exceed 18 months, automatic initiation of the penetration room ventilation system shall be demonstrated.

SR 3.7.11.2 & (LATER) (5.0)

~~4.11.4a The tests and sample analysis of Specification 3.13.1a, b, & c. shall be performed at intervals not to exceed 18 months or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.~~ LATER

~~b. Cold DOP testing shall also be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.~~

~~c. Halogenated hydrocarbon testing shall also be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.~~

SR 3.7.11.1 4.11.5 Each circuit shall be operated at least 1 hour every month. ~~This test shall be considered satisfactory if control board indication verifies that all components have responded properly to the actuation signal.~~ L14 LA3 TRM

5.5.11

4.11 PENETRATION ROOM VENTILATION SYSTEM SURVEILLANCE

<LATER> (3.7)

Applicability

Applies to the surveillance of the penetration room ventilation system. — LATER

Objective

To verify an acceptable level of efficiency and operability of the penetration room ventilation system.

Specification — LA3 VFTP & LATER

5.5.11.d & <LATER> (3.7)

4.11.1 ~~At intervals not to exceed 18 months,~~ the pressure drop across the combined HEPA filters, and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (± 10%). , other filters in the system, — MS

5.5.11 & <LATER> (3.7)

4.11.2 Initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers. — LA3 VFTP & LATER

<LATER> (3.7)

4.11.3 At intervals not to exceed 18 months, automatic initiation of the penetration room ventilation system shall be demonstrated. — LATER

5.5.11 & <LATER> (3.7)

4.11.4a The tests and sample analysis of Specification 3.13.1a, b, & c. shall be performed at intervals not to exceed 18 months or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall also be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

LA3 VFTP & LATER

<LATER> (3.7)

4.11.5 Each circuit shall be operated at least 1 hour every month. This test shall be considered satisfactory if control board indication verifies that all components have responded properly to the actuation signal. — LATER

A2

Bases

The penetration room ventilation system is designed to collect and process potential reactor building penetration room leakage to minimize environmental activity levels resulting from post accident reactor building leaks. The system consists of a sealed penetration room, two redundant filter trains and two redundant fans discharging to the unit vent. The entire system is activated by a reactor building pressure engineered safety features signal and initially requires no operator action.

Since the system is not normally operated, a periodic test is required to show that the system is available for its engineered safety features function. During this test the system will be inspected for such things as water, oil, or other foreign material, gasket deterioration in the HEPA units, and unusual or excessive noise or vibration when the fan motor is running.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 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 18 months to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant and of the HEPA filter bank with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d. of Regulatory Guide 1.52. Radioactive methyl iodide removal efficiency tests shall be performed in accordance with ASTM D3803-1989. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbents qualified according to Regulatory Guide 1.52.

Operation of the system each month for 1 hour will demonstrate operability of the active system components and the filter and adsorber system. If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

LAR A7

4.12 HYDROGEN RECOMBINERS SURVEILLANCE

<u>Applicability</u>

Applies to the surveillance of the hydrogen recombiner systems.

<u>Objective</u>

To verify an acceptable level of efficiency and operability of the hydrogen recombiner systems.

<u>Specification</u>

<LATER> (3.6)

4.12.1 Each hydrogen recombiner system shall be demonstrated OPERABLE: LATER

a. At least once per 6 months by verifying during a 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.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

2. 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

3. Verifying the integrity of the 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.

Table 3.3.15-1 #10
SR 3.3.15.2

4.12.2 Hydrogen concentration instruments shall be calibrated once every 18 months ~~with proper consideration to moisture effect.~~ LA1 Bases

<u>Bases</u>

The OPERABILITY of the recombiners for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the A2

<Add SR 3.3.15.1 for PAM #10> M2

4.12 HYDROGEN RECOMBINERS SURVEILLANCE

~~Applicability~~

~~Applies to the surveillance of the hydrogen recombiner systems.~~ A1

~~Objective~~

~~To verify an acceptable level of efficiency and operability of the hydrogen recombiner systems.~~

SR 3.6.7.1 Specification

4.12.1 Each hydrogen recombiner system shall be demonstrated OPERABLE: L8

a. At least once per ~~6 months~~ 18 Months by verifying during a 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 maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 KW.~~ LA1 Bases

SR 3.6.7.2
SR 3.6.7.3

b. At least once per 18 months by: LA2 TRM

~~1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,~~

2. 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 A13 LA1 Bases

3. Verifying the integrity of the 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.~~ LA1 Bases

<Later> (3.3 D) 4.12.2 Hydrogen concentration instruments shall be calibrated once every 18 months with proper consideration to moisture effect. LATER

~~Bases~~

~~The OPERABILITY of the recombiners for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the~~ A2

3.3.15

expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. The hydrogen recombiner systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following LOCA", Rev. 2, November, 1978.

A2

3.6.7

~~expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. The hydrogen recombiner systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following LOCA", Rev. 2, November, 1978..~~

A2

3.7.8

4.13 EMERGENCY COOLING POND

Applicability

Applies to the emergency cooling pond.

Objective

To verify the availability of a sufficient supply of cooling water inventory in the emergency cooling pond.

A1

Specification

4.13.1 The emergency cooling pond shall be determined operable:

SR 3.7.8.1

1. At least once per 24 hours by verifying the pond's indicated water level is ≥ 5 feet.

SR 3.7.8.2 & Note

2. At least once per 24 hours during the period from June 1 through September 30 by verifying that the pond's average water temperature at the point of discharge from the pond is within its limit.

SR 3.7.8.3

3. At least once per 12 months by making soundings of the pond and verifying an average depth of 5 feet and that the contained water volume of the pond is within its limit.

LA1 Bases

SR 3.7.8.3

4. At least once per 12 months by a visual inspection of the loose stone (riprap) placed on the banks of the pond and of the concrete slab spillway and verifying that the earth portions of the stone covered embankments and the spillway:

   1. Have not been eroded or undercut by wave action, and

   2. Do not show apparent changes in visual appearance or other abnormal degradation from their as built condition.

LA1 Bases

Bases

The requirements of Specification 4.13 provide for verification of a sufficient water inventory in the emergency cooling pond to handle a DBA with a concurrent failure of the Dardanelle Reservoir. This specification ensures that Specification 3.11.1 is met. Monitoring temperature only during the period June 1 through September 30 of each year ensures that, during the hot summer months, the pond temperature limit is not exceeded. During other periods of the year the pond temperature will not have the potential to reach the temperature limit. Soundings are performed to ensure the water volume is within limits and that the indicated level is indicative of an equivalent water volume for accident mitigation. The measured ECP temperature at the discharge from the pond is considered a conservative average of total pond conditions since solar gain, wind speed, and thermal current effects throughout the pond will essentially be at equilibrium conditions under initial stagnant conditions. Visual inspections are performed to ensure any physical degradation is within acceptable limits to enable the ECP to fulfill its safety function. An engineering evaluation sr be performed by a qualified engineer of any apparent changes in visual appearance or other abnormal degradation to determine operability.

A2

3.7

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4.14 RADIOACTIVE MATERIALS SOURCES SURVEILLANCE

Applicability

Applies to leakage testing of byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

Test for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

1. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferrer indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Each sealed startup source shall be leak tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

4. The periodic leak test does not apply to the boronometer source. This source shall be tested for leakage at least once per 18 months.

Items on this page also addressed in the following packages: NA

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~~4.15 AUGMENTED INSERVICE INSPECTION PROGRAM FOR HIGH ENERGY LINES OUTSIDE OF CONTAINMENT~~

~~Applicability~~

~~Applies to welds in piping systems located outside of containment where protection from the consequences of postulated ruptures is not provided by a system of pipe whip restraints, jet impingement barriers, protective enclosures and/or other measures designed specifically to cope with such ruptures.~~

~~For Arkansas Nuclear One-Unit 1 this specification applies to six welds in the main steam and main feedwater lines identified as welds 6, 7, 23, 24, 55, and 56 on Figures A-7, A-8 and A-15 of the Final Safety Analysis Report.~~

~~Objective~~

~~To provide assurance of the continued integrity of the piping system over their service lifetime.~~

~~Specifications~~

~~4.15.1 At the first refueling outage period, a volumetric examination shall be performed with 100 percent inspection of each weld in accordance with the requirements of ASME Code Section XI, Inservice Inspection of Nuclear Power Plant Components, to establish system integrity and baseline data.~~

~~4.15.2 The inservice inspection at each weld shall be performed in accordance with the requirements of ASME Code Section XI, Inservice Inspection of Nuclear Power Plant Components, with the following schedule:~~

~~(The inspection intervals identified below sequentially follow the baseline examination of 4.15.1).~~

~~First Inspection Interval~~

~~a. First 3-1/3 years (or nearest refueling outage)~~	~~100% volumetric inspection of each weld~~
~~b. Second 3-1/3 years (or nearest refueling outage)~~	~~100% volumetric inspection of each weld~~
~~c. Third 3-1/3 years (or nearest refueling outage)~~	~~100% volumetric inspection of each weld~~

Items on this page also addressed in the following packages NA

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Successive Inspection Intervals

Every 10 years thereafter (or nearest refueling outage)	Volumetric inspection of two of the welds at the expiration of each 1/3 of the inspection intervals with a cumulative 100% coverage of all welds.

Note - The welds selected during each inspection period shall be distributed among the total number to be examined to provide a representative sampling of the conditions of the welds.

4.15.3 In the event repairs of any welds are required following any examination during successive inspection intervals, the inspection schedule for the repaired welds will revert back to the first 10 year inspection program.

4.15.4 Examinations that reveal unacceptable structural defects in a weld during an inspection under 4.15.2 should be extended to require an additional inspection of another 1/3 of the welds. If further unacceptable defects are detected in the second sampling, the remainder of the welds shall be inspected.

4.15.5 Repairs, reexamination and piping pressure tests shall be conducted in accordance with Section XI of the ASME Code.

Items on this page also addressed in the following packages: N A

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4.16 SHOCK SUPPRESSORS (Snubbers)

Applicability

This technical specification applies to all shock suppressors (snubbers). The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

Objective

Verify an acceptable level of operability of the shock suppressors protecting the primary system and any other safety related system or component.

Specification

4.16.1 The following surveillance requirements apply to all applicable shock suppressors.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers may be categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.16-1. The visual inspection interval for each category of snubber shall be determined based upon the criteria provided in Table 4.16.1.

Items on this page also addressed in the following packages: NA

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(R) TRM

c. <u>Visual Inspection Acceptance Criteria</u>

Visual inspections shall verify (1) that there are no visible indications of damage or impaired operability, and (2) attachments to the foundation or supporting structure are functional and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as INOPERABLE and may be reclassified OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined operable per Specifications 4.16.1.d or 4.16.1.e, as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined operable via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to a common hydraulic fluid reservoir shall be evaluated for operability if any snubber connected to that reservoir is determined to be inoperable.

d. <u>Functional Tests</u>

At least once each refueling shutdown a representative sample of snubbers shall be tested using the following sample plan.

At least 10% of the snubbers required by Specification 3.16.1 shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.16.1.e, an additional 10% of the snubbers shall be functionally tested until no more failures are found or until all snubbers have been functionally tested.

The representative samples for the functional test sample plans shall be randomly selected from the snubbers required by Specification 3.16.1 and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of sizes, and capacities. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

Items on this page also addressed in the following packages: NA

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e. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both direction of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

f. Functional Test Failure Analysis

An evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the operability of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to activate or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or

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design deficiency, all snubbers of the same type subject to the same defect shall be evaluated in a manner to ensure their operability. This testing requirement shall be independent of the requirements stated in Specification 4.16.1.d for snubbers not meeting the functional test acceptance criteria.

g. Preservice Testing of Repaired, Replacement and New Snubbers

Preservice operability testing shall be performed on repaired, replacement or new snubbers prior to installation. Testing may be at the manufacturer's facility. The testing shall verify the functional test acceptance criteria in Specification 4.16.1.e.

In addition, a preservice inspection shall be performed on each repaired, replacement or new snubber and shall verify that:

1) There are no visible signs of damage or impaired operability as a result of storage, handling or installation;

2) The snubber load rating, location, orientation, position setting and configuration (attachments, extensions, etc.), are in accordance with design;

3) Adequate swing clearance is provided to allow snubber movement;

4) If applicable, fluid is at the recommended level and fluid is not leaking from the snubber system;

5) Structural connections such as pins, bearings, studs, fasteners and other connecting hardware such as lock nuts, tabs, wire and cotter pins are installed correctly.

h. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The expected service life for the various seals, seal materials, and applications shall be determined and established based on engineering information and the seals shall be replaced so that the expected service life will not be exceeded during a period when the snubber is required to be operable. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.9.2.

Items on this page also addressed in the following packages: NA

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TABLE 4.16-1
SNUBBER VISUAL INSPECTION INTERVAL

NUMBER OF INOPERABLE SNUBBERS

Population per Category (Notes 1 and 2)	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber category shall be determined based upon the previous inspection interval and the number of INOPERABLE snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, categories must be determined and documented before any inspection and that determination shall be the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population per category and the number of INOPERABLE snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, and C if that integer includes a fractional value of INOPERABLE snubbers as determined by interpolation.

Items on this page also addressed in the following packages: NA

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TABLE 4.16-1 (Continued)
SNUBBER VISUAL INSPECTION INTERVAL

Note 3: If the number of INOPERABLE snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of INOPERABLE snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of INOPERABLE snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of INOPERABLE snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of INOPERABLE snubbers found during the previous interval and the number in Column B to the difference in the numbers in Column B and C.

Note 6: Specified surveillance intervals may be adjusted plus or minus 25 percent to accommodate normal test and surveillance schedule intervals up to and including 48 months, with the exception that inspection of inaccessible snubbers may be deferred to the next shutdown when plant conditions allow five days for inspection. See Note 7 for definition of interval as applied to snubber visual inspections. The provisions of Specification 4 regarding surveillance intervals are not applicable.

Note 7: Interval as defined for the shock suppressors (snubbers) visual inspection surveillance requirements is the period of time starting when the unit went into cold shutdown for refueling, and ending when the unit goes into cold shutdown for its next scheduled refueling. This period of time is nominally considered to be an 18 month period, or a 24 month period based on the type of fuel being used. However, the period of time (interval) could be shorter or longer due to plant operating variables such as fuel life and operating performance.

Items on this page also addressed in the following packages: NA

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BASES

All safety related snubbers are required to be operable to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure, or failure of the system on which they are installed, would have no adverse effect on any safety related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to plant systems. Therefore, the required inspection interval varies based upon the number of INOPERABLE snubbers found during the previous inspection in proportion to the sizes of the various snubber populations or categories and the previous inspection interval as specified in NRC Generic Letter 90-09, "Alternative Requirements For Snubber Visual Inspection Intervals and Corrective Actions". Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the result of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety related component or system has been adversely affected by inoperability of the snubber. The engineering evaluation is performed to determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

If a review and evaluation of an INOPERABLE snubber is performed and documented to justify continued operation, and provided that all design criteria are met with the INOPERABLE snubber, then the INOPERABLE snubber would not need to be restored or replaced.

3.7.12

<Add SR 3.7.12.1> M25

4.17 FUEL HANDLING AREA VENTILATION SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the fuel handling area ventilation system. A1

Objective

To verify an acceptable level of efficiency and operability of the fuel handling area ventilation system.

Specification

SR 3.7.12.2 & <LATER> (S.0)

4.17.1 At intervals not to exceed 18 months, pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%). LATER

4.17.2 Initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers.

4.17.3 a. The tests and sample analysis of Specification 3.15.1.a, b, & c shall be performed within 720 system operating hours prior to irradiated fuel handling operations in the auxiliary building, and prior to irradiated fuel handling in the auxiliary building following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing.

4.17.4 The system shall be operated for at least 10 hours prior to initiation of irradiated fuel handling operations in the auxiliary building if it has not been operated for at least 10 hours within the previous 30 days. L14

Bases

Since the fuel handling area ventilation system may be in operation when fuel is stored in the pool but not being handled, its operability must be verified before handling of irradiated fuel. Operation of the system for 10 hours before irradiated fuel handling operations and performance of Specification 4.17.3 will demonstrate operability of the active system components and the filter and adsorber systems. A2

5.5.11

4.17 FUEL HANDLING AREA VENTILATION SYSTEM SURVEILLANCE

⟨LATER (3.7)⟩ — LATER

Applicability

Applies to the surveillance of the fuel handling area ventilation system.

Objective

To verify an acceptable level of efficiency and operability of the fuel handling area ventilation system.

Specification

5.5.11 5.5.11.d & LATER (3.7)

4.17.1 ~~At intervals not to exceed 18 months,~~ pressure drop across the combined HEPA filters, and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%). , other filters in the system, — LA3 VFTP & LATER — M5

5.5.11 & LATER (3.7)

4.17.2 Initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers. — LA3 VFTP & LATER

5.5.11 & LATER (3.7)

4.17.3 a. The tests and sample analysis of Specification 3.15.1.a, b, & c shall be performed within 720 system operating hours prior to irradiated fuel handling operations in the auxiliary building, and prior to irradiated fuel handling in the auxiliary building following significant painting, fire or chemical release in any ventilation zone communicating with the system. — LA3 VFTP & LATER

b. Cold DOP testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing.

⟨LATER (3.7)⟩

4.17.4 The system shall be operated for at least 10 hours prior to initiation of irradiated fuel handling operations in the auxiliary building if it has not been operated for at least 10 hours within the previous 30 days. — LATER

Bases

Since the fuel handling area ventilation system may be in operation when fuel is stored in the pool but not being handled, its operability must be verified before handling of irradiated fuel. Operation of the system for 10 hours before irradiated fuel handling operations and performance of Specification 4.17.3 will demonstrate operability of the active system components and the filter and adsorber systems. — A2

3.7.12

A2

~~Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 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 and air distribution should be determined once every 18 months to show system performance capability.~~

~~The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant and of the HEPA filter bank with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d. of Regulatory Guide 1.52. Radioactive methyl iodide removal efficiency tests shall be performed in accordance with ASTM D3803-1989. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbents qualified according to Regulatory Guide 1.52.~~

LAR A7

5.5.9

5.5.9 4.18 STEAM GENERATOR (SG) TUBE ~~TUBING~~ SURVEILLANCE PROGRAM

~~Applicability~~

~~Applies to the surveillance of tubing of each steam generator.~~ A1

~~Objective~~

This program provides controls to ensure the A1

~~To ensure~~ integrity of the steam generator tubing through a defined inservice surveillance program, and to minimize exposure of personnel to radiation during performance of the surveillance program.

<INSERT 110jA> A1

~~Specification~~

~~4.18.1 Baseline Inspection~~

a. The first steam generator tubing inspection performed ~~according to Specifications 4.18.2 and 4.18.3.a~~ 5.5.9.b and 5.5.9.c.1 shall be considered as constituting the baseline condition for subsequent inspections. in accordance with

b. ~~4.18.2~~ Examination Methods:

1. Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness except for a sleeved tube at the lower sleeve end. A1

2. For examination of the sleeved steam generator tubing at the lower sleeve end, the indications will be compared to those obtained during the baseline sleeved tube inspection. Significant deviations between these indications will be considered sufficient evidence to warrant designation as a degraded tube. Direct quantification of the 40 percent through-wall plugging limit is available with eddy-current testing.

c. ~~4.18.3~~ Selection and Testing

The steam generator sample size is specified in Table ~~4.18-1~~ 5.5.9-1. The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table ~~4.18-2~~ 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in ~~Specification 4.18.4~~ 5.5.9.d and the inspected tubes shall be verified acceptable per the acceptance criteria of ~~Specification 4.18.5~~ 5.5.9.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators; the tubes selected for these inspections shall be selected on a random basis except: A1

<CTS INSERT 110jA>

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the SG Tube Surveillance Program inspection frequencies.

5.5.9

<All changes> — A1

1 ~~1.~~ The first sample inspection during each inservice inspection (subsequent to the baseline inspection) of each steam generator shall include:

i ~~1.~~ All nonplugged tubes that previously had detectable wall penetrations (>20%), except tubes in which the wall penetration has been spanned by a sleeve, and

ii ~~2.~~ At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems, except where specific groups are inspected per ~~Specification 4.18.3.a.8.~~ 5.5.9.c.1.iii

A tube inspection (pursuant to ~~Specification 4.18.5.a.9~~ 5.5.9.e.1.ix) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

iii ~~3.~~ Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. The inspection may be concentrated on those portions of the tubes where imperfections were previously found. No credit will be taken for these tubes in meeting minimum sample size requirements. Where only a portion of the tube is inspected, the remainder of the tube will be subjected to the random inspection.

(1) Group A-1: Tubes within one, two or three rows of the open inspection lane.

(2) Group A-2: Unplugged tubes with sleeves installed.

(3) Group A-3: Tubes in the wedge-shaped group on either side of the lane region (Group A-1) as defined by Figure ~~4.18.1~~ 5.5.9-1

iv ~~4.~~ Tubes with axially-oriented tube end cracks (TEC) which have been left inservice for the previous cycle shall be inspected with a rotating coil eddy current technique in the area of the TEC and characterized in accordance with topical report BAW-2346P, Rev.0, during all subsequent SG inspection intervals pursuant to ~~4.18.4.~~ 5.5.9.d The results of this examination may be excluded from the first random sample. Tubes with axial TECs identified during previous inspections which meet the criteria to remain in service will not be included when calculating the inspection category of the OTSG.

v ~~5.~~ Implementation of the upper tubesheet ODIGA alternate repair criteria requires a 100% bobbin coil inspection of the non-plugged and non-sleeved tubes, spanning the defined region of the upper tubesheet, during all subsequent SG inspection intervals pursuant to ~~4.18.4.~~ 5.5.9.d Tubes with ODIGA identified during previous inspections which meet the criteria to remain in service will not be included when calculating the inspection category for the OTSG. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. ODIGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes in accordance with topical report BAW-10235P, Revision 1.

5.5.9

2. ~~5~~. All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.

3. ~~6~~. The second and third sample inspections during each inservice inspection as required by Table 5.5.9-2 ~~4.18-2~~ may be less than a full tube inspection by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found. (A1)

4. The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected, are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES: (1) In all inspections, previously degraded tubes whose degradations have not been spanned by a sleeve must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 5.5.9.c.1.iii ~~4.18.3.a.3~~ defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

(3) Where special inspections are performed pursuant to 5.5.9.c.2 ~~4.18.3.b~~ defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found above the new roll area, are not included in the determination for the inspection results category of a general steam generator inspection. (A1)

5.5.9

(ALL CHANGES) A1

~~4.18.4 Inspection Intervals~~

d. The above-required inservice inspections of steam generator tubes shall be performed at the following frequencies:

1. ~~a.~~ The baseline inspection shall be performed during the first refueling shutdown. Subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group* of tubes following service under all volatile treatment (AVT) conditions fall into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval for that group may be extended to a maximum of 40 months.

2. ~~b.~~ If the results of the inservice inspection of a steam generator performed in accordance with Table ~~4.18-2~~ 5.5.9-2 at 40-month intervals for a given group* of tubes fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 20 calendar months after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in ~~4.18.4.a~~ 5.5.9.d.1 and the interval can be extended to 40 months.

3. ~~c.~~ Additional unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table ~~4.18-2~~ 5.5.9-2 during the shutdown subsequent to any of the following conditions:

   i. ~~1.~~ Primary-to-secondary leakage in excess of the limits of Specification ~~3.1.6.3.b~~ 3.4.13 (Inservice inspection not required if leaks originate from tube-to-tubesheet welds). If the leaking tube is from either Group A-1 or A-3 as defined in Specification ~~4.18.3.a.3~~ 5.5.9.c.1.iii, all of the tubes in the affected group in this steam generator may be inspected in lieu of the first sample inspection specified in Table ~~4.18-2~~ 5.5.9-2. If the degradation mechanism which caused the leak is limited to a specific portion of the tube length, the inspection per this paragraph may be limited to the affected portion of the tube length. If the results of this inspection fall into the C-3 category, all of the tubes in the same group in the other steam generator will also be similarly inspected.

   If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all of the tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, all of the tubes with rerolled areas in the other steam generator will also be similarly inspected. This inspection will be in lieu of the first sample inspection specified in Table ~~4.18-2~~ 5.5.9-2.

   ii. ~~2.~~ A seismic occurrence greater than the Operating Basis Earthquake,

   iii. ~~3.~~ A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   iv. ~~4.~~ A main steam line or feedwater line break.

*A group of tubes means: (a) All tubes inspected pursuant to ~~4.18.3.a.3~~ 5.5.9.c.1.iii or
(b) All tubes in a steam generator less those inspected pursuant to ~~4.18.3.a.3~~ 5.5.9.c.1.iii

5.5.9

e. ~~4.18.5~~ Acceptance Criteria

1 ~~a.~~ As used in this specification:

i ~~1.~~ Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

ii ~~2.~~ Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

iii ~~3.~~ Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.

iv ~~4.~~ Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation, except where all degradation has been spanned by the installation of a sleeve.

v ~~5.~~ % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

vi ~~6.~~ Defect means an imperfection of such severity that it exceeds the plugging limit except where the imperfection has been spanned by the installation of a sleeve. A tube containing a defect in its pressure boundary is defective.

vii ~~7.~~ Plugging Limit means the imperfection depth at or beyond 40% of the nominal tube wall thickness for which the tube shall be sleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. This does not apply during Cycle 16 to ODIGA indications within the defined region of the upper tubesheet. These indications shall be assessed for continued plant operation in accordance with topical report BAW-10235P, Revision 1.

Axially-oriented TEC indications in the tube that do not extend beyond the adjacent cladding portion of the tube sheet into the carbon steel portion are not included in this definition. These indications shall be assessed for continued plant operation in accordance with topical report BAW-2346P, Rev. 0.

The reroll repair process will only be used to repair tubes with defects in the upper tubesheet area. The reroll repair process will be performed only once per steam generator tube using a 1 inch roll length. The new roll area must be free of detectable degradation in order for the repair to be considered acceptable. The reroll repair process is described in the topical report, BAW-10232P, Revision 00.

viii ~~8.~~ Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, ~~or a steam line or feedwater line~~ break as specified in ~~Specification 4.18.4.c~~ 5.5.9.d.3 (A1)

ix ~~9.~~ Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. For tubes that have been repaired by the reroll process within the upper tubesheet, that portion of the tube above the new roll can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

SR 3.4.13.2 & <LATER> (5.0)

b. The steam generator shall be determined operable ~~after completing the corresponding actions (plug, reroll, or sleeve all tubes exceeding the plugging limit and all tubes containing non-TEC through-wall cracks) required by Table 4.18-2.~~

A5 & LATER

Tube 110/60 in the "A" steam generator contains indications in the upper roll transition that exceed the plugging limit. Tube 110/60 may remain in service with these indications for the duration of cycle 16 without rendering the "A" steam generator inoperable.

4.18.6 Reports

Following each inservice inspection of steam generator tubes, the complete results of the inspection shall be reported to the NRC. This report, to be submitted within 90 days of inspection completion, shall include:

a. Number and extent of tubes inspected;

b. Location and percent of wall-thickness penetration for each indication of an imperfection;

c. Identification of tubes plugged and tubes sleeved;

d. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes;

e. Summary of the condition monitoring and operational assessment results when applying TEC alternate repair criteria; and

f. Summary of the condition monitoring and the operational assessment results (including growth) when applying the upper tubesheet ODIGA alternate repair criteria.

This report shall be in addition to a Special Report (per Specification 6.12.5.d) required for the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 4.18-2. The Commission shall be notified of the results of steam generator tube inspections which fall into Category C-3 prior to resumption of plant operation. The written Special Report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.5.9
5.6.7

unless otherwise noted: < ALL CHANGES = A1 >

2 ~~b~~. The steam generator shall be determined operable after completing the corresponding actions (plug, reroll, or sleeve all tubes exceeding the plugging limit and all tubes containing non-TEC through-wall cracks) required by Table ~~4.18-2~~ 5.5.9-2

~~Tube 110/60 in the "A" steam generator contains indications in the upper roll transition that exceed the plugging limit. Tube 110/60 may remain in service with these indications for the duration of cycle 16 without rendering the "A" steam generator inoperable.~~ A16

5.6.7 ~~4.18.6~~ ~~Reports~~ Steam Generator Tube Surveillance

Following each inservice inspection of steam generator tubes, the complete results of the inspection shall be reported to the NRC. This report, to be submitted within 90 days of inspection completion, shall include:

a. Number and extent of tubes inspected;

b. Location and percent of wall-thickness penetration for each indication of an imperfection;

c. Identification of tubes plugged and tubes sleeved;

d. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes;

e. Summary of the condition monitoring and operational assessment results when applying TEC alternate repair criteria; and

f. Summary of the condition monitoring and the operational assessment results (including growth) when applying the upper tubesheet ODIGA alternate repair criteria.

~~This report shall be in addition to a Special Report (per Specification 6.12.5.d) required for the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 4.18-2.~~ A4 The Commission shall be notified of the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 5.5.9-2 prior to resumption of plant operation. The written Special Report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.5.9

A2

Bases

The surveillance requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

In general, steam generator tubes that are degraded beyond the repair limit can either be plugged, sleeved, or rerolled. The steam generator (SG) tubes that are plugged are removed from service by the installation of plugs at both ends of the associated tube and thus completely removing the tube from service. When the tube end cracking (TEC) alternate repair criteria is applied, axially-oriented indications found not to extend from the tube sheet cladding region into the carbon steel region may be left in service under the guidelines of topical report BAW-2346P, Rev. 0. When the upper tubesheet outer diameter intergranular attack (ODIGA) alternate repair criteria is applied, indications found within the defined region of the upper tubesheet may be left in service under the guidelines of topical report BAW-10235P, Revision 1. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. Following a SG inspection, an operational assessment is performed to ensure primary-to-secondary leak rates will be maintained within the assumptions of the accident analysis.

Degraded steam generator tubes can also be repaired by the installation of sleeves which span the area of degradation and serve as a replacement pressure boundary for the degraded portion of the tube, thus permitting the tube to remain in service.

Degraded steam generator tubes can also be repaired by the rerolling of the tube in the upper tubesheet to create a new roll area and pressure boundary for the tube. The rerolling methodology establishes a new pressure boundary below the degradation, thus permitting the tube to remain in service. The degraded tube above the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed in the upper tubesheet. The rerolling repair process will only be used to repair defects in the upper tubesheet in accordance with BAW-10232P, Revision 00.

All tubes which have been repaired using the reroll process will have the new roll area inspected during future inservice inspections. Defective or degraded tube indications found in the new roll and any indications found in the original roll need not be included in determining the Inspection Results Category for the generator inspection.

The reroll repair process will only be used to repair tubes with defects in the upper tubesheet area. The reroll repair process will be performed only once per steam generator tube using a 1 inch roll length. Thus, multiple applications of the reroll process to any individual tube is not acceptable. The new roll area must be free of detectable degradation in order for the repair to be considered acceptable. After the new roll area is initially deemed acceptable, future degradation in the new roll area will be analyzed to determine if the tube is defective and needs to be removed from service. The reroll repair process is described in the topical report, BAW-10232P, Revision 00.

5.5.9

TABLE ~~4.18-1~~ 5.5.9-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No
No. of Steam Generators per Unit	Two
First Inservice Inspection	Two
Second & Subsequent Inservice Inspection	One[1]

Table Notation:

1 The inservice inspection may be limited to one steam generator on alternating schedule encompassing 3N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE ~~4.18-2~~ 5.5.9-2 A1

STEAM GENERATOR TUBE INSPECTION [2,3]

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.[1]	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug, reroll, or sleeve defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug, reroll, or sleeve defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug, reroll, or sleeve defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug, reroll, or sleeve defective tubes and inspect 2S tubes in other S.G. ~~Special Report to NRC pursuant to 6.12.5.d~~	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S.G. and plug, reroll, or sleeve defective tubes. ~~Special Report to NRC pursuant to 6.12.5.d~~	N/A	N/A

A4

NOTES: [1] $S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

[2] For tubes inspected pursuant to ~~4.18.3.a.3~~ 5.5.9.c.1.iii: No action is required for C-1 results. For C-2 results in one or both steam generators plug, reroll, or sleeve defective tubes. For C-3 results in one or both steam generators, plug, reroll, or sleeve defective tubes and provide a Special Report to NRC pursuant to ~~6.12.5.d~~ 5.6.7

[3] No more than ten thousand (10,000) sleeves may be installed in both ANO-1 steam generators combined.

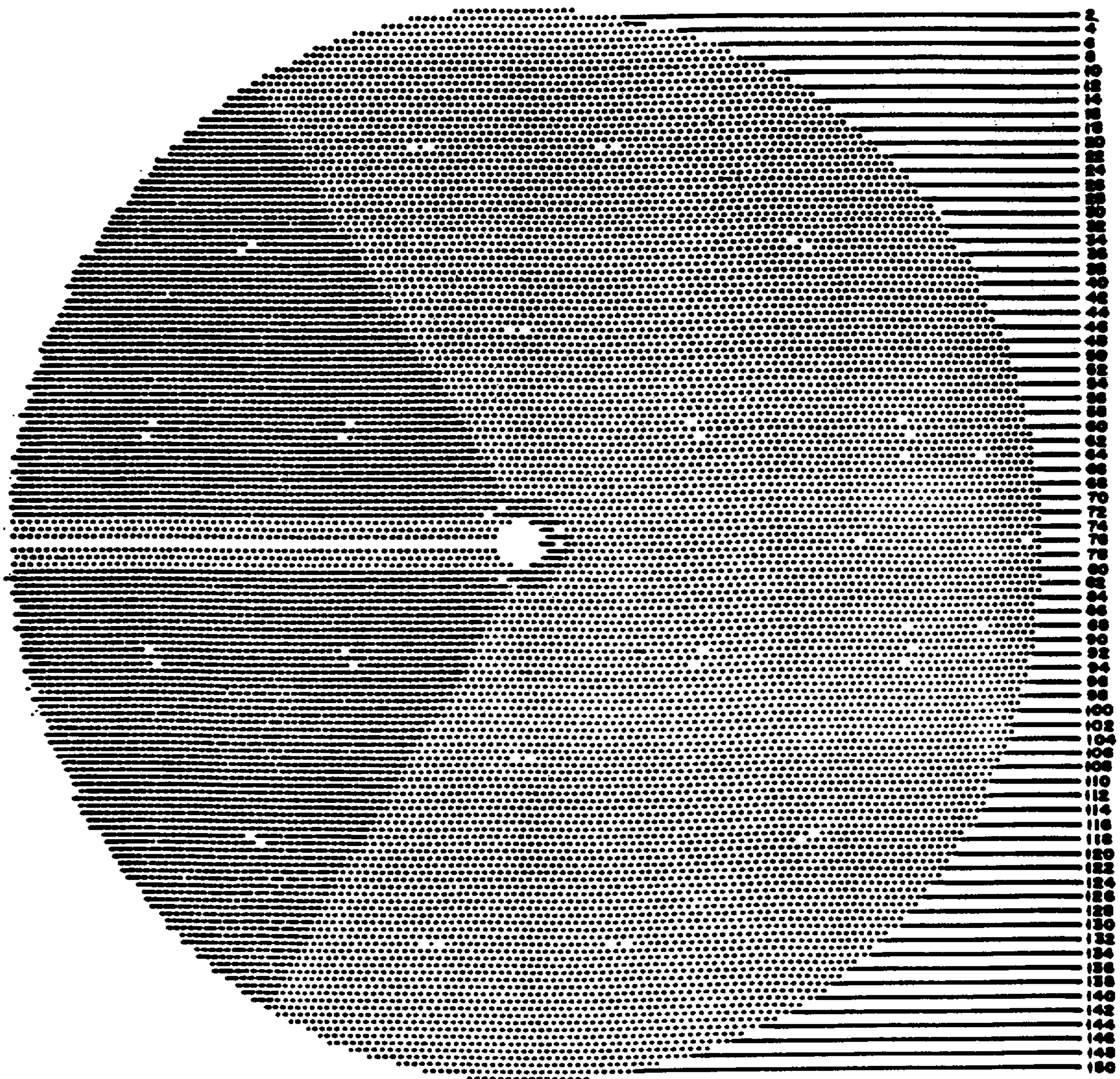
A1

5.5.9

Figure ~~4.18.1~~ 5.5.9-1

Upper Tube Sheet View of Wedge Shaped Group (Group A-3) per Specification ~~4.18.3.a.3~~ 5.5.9.c.1.iii

(A1)



DESCRIPTION	TUBE COUNT
GROUP A - 1: Lane region tubes as defined in ~~4.18.3.a.3(1)~~ 5.5.9.c.1.iii(1)	382
GROUP A - 3: Wedge shaped group depicted by darkened region of figure	4880

(A1)

Items on this page also addressed in the following packages: NA

5R

R TRM

4.25 <u>Reactor Building Purge Filtration System</u>

<u>Applicability</u>

Applies to the surveillance of the reactor building purge filtration system.

<u>Objective</u>

To verify an acceptable level of efficiency and operability of the reactor building purge filtration system.

<u>Specification</u>

4.25.1 The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%) within 720 system operating hours prior to initial irradiated fuel handling operations.

4.25.2 Initially and after any maintenance or testing that could affect the air distribution within the reactor building purge system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers.

4.25.3a. The tests and sample analysis of Specification 3.22.1.a, b, & c. shall be performed within 720 system operating hours prior to initial irradiated fuel handling operations in the reactor building, and prior to irradiated fuel handling in the reactor building following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall also be performed prior to irradiated fuel handling in the reactor building after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed prior to irradiated fuel handling in the reactor building after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing.

<u>Bases</u>

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 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 refueling period to show system performance capability.

Items on this page also addressed in the following packages: NA

SR

R TRM

~~The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant and of the HEPA filter bank with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d. of Regulatory Guide 1.52. Radioactive methyl iodide removal efficiency tests shall be performed in accordance with RDT Standard M16-IT. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbents qualified according to Regulatory Guide 1.52.~~

3.6.3

4.26 REACTOR BUILDING PURGE VALVES

Applicability

~~This specification applies to the reactor building purge supply and exhaust isolation valves.~~ A1

Objective

~~To assure reactor building integrity.~~

LA1 Bases

Specification

SR 3.6.3.1 4.26.1 The reactor building purge ~~supply and exhaust~~ isolation valves shall be determined closed at least once per 31 days ~~when containment integrity is required by TS 3.6.1.~~ during MODES 1, 2, 3 and 4 M1

<Later> (5.0) 4.26.2 ~~Prior to exceeding conditions which require establishment of reactor building integrity per TS 3.6.1, the leak rate of the purge supply and exhaust isolation valves shall be verified to be within acceptable limits per TS 4.4.1, unless the test has been successfully completed within the last three months.~~ LATER

Bases A2

~~Determination of reactor building purge valve closure will ensure that reactor building integrity is not unintentionally breached.~~

~~As a result of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," it was concluded that excess leakage past valve resilient seals is typically caused by severe environmental conditions and/or wear due to use. Recommended leak test frequencies of three months are deemed to be adequate to detect seal degradation of resilient seals.~~

~~The three month test need not be conducted with the precision of the Type C 10CFR50, Appendix J criteria, however the test must be sufficient to detect degradation.~~

5.5.16

4.26 REACTOR BUILDING PURGE VALVES

Applicability

This specification applies to the reactor building purge supply and exhaust isolation valves.

Objective

To assure reactor building integrity.

Specification

<LATER> (3.6) 4.26.1 The reactor building purge supply and exhaust isolation valves shall be determined closed at least once per 31 days when containment integrity is required by TS 3.6.1. LATER

5.5.16 4.26.2 Prior to exceeding conditions which require establishment of reactor building integrity per TS 3.6.1, the leak rate of the purge supply and exhaust isolation valves shall be verified to be within acceptable limits per TS 4.4.1, unless the test has been successfully completed within the last three months.

A1

M8

Bases

Determination of reactor building purge valve closure will ensure that reactor building integrity is not unintentionally breached.

As a result of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," it was concluded that excess leakage past valve resilient seals is typically caused by severe environmental conditions and/or wear due to use. Recommended leak test frequencies of three months are deemed to be adequate to detect seal degradation of resilient seals.

The three month test need not be conducted with the precision of the Type C 10CFR50 Appendix J criteria, however the test must be sufficient to detect degradation.

A2

3.4.5
3.4.6
3.4.7
3.4.8

4.27 DECAY HEAT REMOVAL

~~APPLICABILITY~~

~~Applies to surveillance of the decay heat removal system and to the reactor coolant loops and associated reactor coolant pumps as needed for decay heat removal.~~

~~OBJECTIVE~~

~~To assure the operability of the decay heat removal system and the reactor coolant loops as needed for decay heat removal.~~

A1

SPECIFICATION

SR 3.4.5.2
SR 3.4.6.2

4.27.1 The required reactor coolant pumps shall be determined operable once per seven (7) days by verifying correct breaker alignments and indicated power availability.

<LATER> (5.0)

~~4.27.2 The required decay heat removal loop(s) shall be determined operable per Specification 4.2.2.~~ LATER

SR 3.4.7.2
3.4.7 LCO #b

4.27.3 The required steam generator(s) shall be determined operable ~~by verifying the secondary side water level to be ≥20 inches in the startup range~~ at least once per 12 hours. L8

SR 3.4.5.1
SR 3.4.6.1

4.27.4 The required reactor coolant loop(s) shall be determined ~~operable by verifying the required loop(s)~~ to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours. A1

SR 3.4.6.1
SR 3.4.7.1
SR 3.4.8.1
& <LATER> (3.9)

4.27.5 The required decay heat removal loop shall be determined to be in operation at least once per 12 hours. & LATER

<Add SR 3.4.7.3 with Note & SR 3.4.8.2 with Note> M13

3.9.4
3.9.5

4.27 DECAY HEAT REMOVAL

APPLICABILITY

Applies to surveillance of the decay heat removal system and to the reactor coolant loops and associated reactor coolant pumps as needed for decay heat removal. —LATER

OBJECTIVE

<LATER> (3.4A)

To assure the operability of the decay heat removal system and the reactor coolant loops as needed for decay heat removal.

SPECIFICATION

<LATER> (3.4A) — 4.27.1 The required reactor coolant pumps shall be determined operable once per seven (7) days by verifying correct breaker alignments and indicated power availability. —LATER

<LATER> (5.0) — 4.27.2 The required decay heat removal loop(s) shall be determined operable per Specification 4.8.2. —LATER

4.27.3 The required steam generator(s) shall be determined operable by verifying the secondary side water level to be ≥20 inches on the startup range at least once per 12 hours. —LATER

<LATER> (3.4A) — 4.27.4 The required reactor coolant loop(s) shall be determined operable by verifying the required loop(s) to be in operation and circulating reactor coolant at least once per 12 hours.

SR 3.9.4.1 & <LATER> (3.4A) — 4.27.5 The required decay heat removal loop shall be determined to be in operation at least once per 12 hours. — & LATER

<Add SR 3.9.5.1 > — M1

5.0

<LATER> (3.4A)

LATER

4.27 DECAY HEAT REMOVAL

APPLICABILITY

Applies to surveillance of the decay heat removal system and to the reactor coolant loops and associated reactor coolant pumps as needed for decay heat removal.

OBJECTIVE

To assure the operability of the decay heat removal system and the reactor coolant loops as needed for decay heat removal.

SPECIFICATION

4.27.1 The required reactor coolant pumps shall be determined operable once per seven (7) days by verifying correct breaker alignments and indicated power availability.

4.27.2 The required decay heat removal loop(s) shall be determined operable per Specification 4.2.2.

A5

<LATER> (3.4A)

4.27.3 The required steam generator(s) shall be determined operable by verifying the secondary side water level to be ≥20 inches on the startup range at least once per 12 hours.

4.27.4 The required reactor coolant loop(s) shall be determined operable by verifying the required loop(s) to be in operation and circulating reactor coolant at least once per 12 hours.

LATER

<LATER> (3.4A & 3.9)

4.27.5 The required decay heat removal loop shall be determined to be in operation at least once per 12 hours.

LATER

5.5.12

4.28 EXPLOSIVE GAS MIXTURE

LA5

EG&STEMP

<u>Applicability</u>

Applies to the Waste Gas System hydrogen/oxygen analyzers.

<u>Objective</u>

To prevent accumulation of explosive mixtures in the waste gas system.

<u>Specification</u>

4.28.1 The concentration of hydrogen/oxygen in the waste gas system shall be monitored continuously by either the primary or redundant waste gas analyzer during waste gas compressing operations to the waste gas decay tanks.

4.28.2 During waste gas system operation, with no H_2/O_2 analyzer in service, without delay suspend all additions of waste gas to the decay tanks or take grab samples for analysis every 4 hours during degassing operations, daily during other operations. The analysis of these samples shall be completed within 8 hours of taking the sample.

<u>Bases</u>

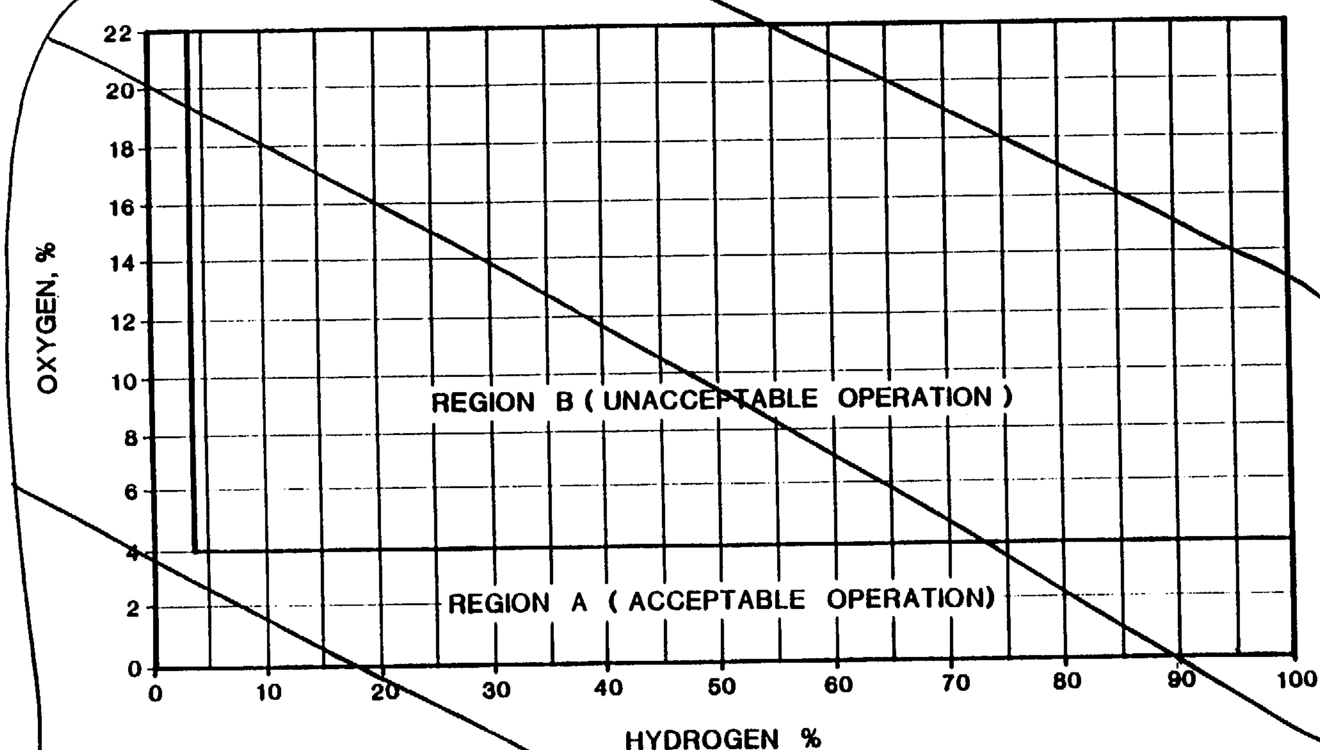
A2

This specification is to assure that the hydrogen/oxygen concentration will be kept within the limits in Figure 3.24-1 and therefore not enter the flammable region concentrations in the waste gas decay tanks.

Grab samples are to be taken every 4 hours during degassing operations when both hydrogen/oxygen analyzers are out of service. These samples are to be analyzed within 8 hours to assure that the hydrogen/oxygen concentration is within the limits in Figure 3.24-1. During other Waste Gas compressor operations, the hydrogen/oxygen concentration is not as subject to change, therefore grab samples are to be taken every 24 hours.

Amendment No. 93

110bc



HYDROGEN - OXYGEN LIMITS FOR ANO - 1 WASTE GAS SYSTEM

FIGURE 3.2 4-1

LA5

EGASTRMP

5.5.12

5.5.12

LA5
EG&SJRM?

4.29 RADIOACTIVE EFFLUENTS

4.29.1 Radioactive Liquid Holdup Tanks

Applicability: At all times

Objective: To ensure that the limits of 10 CFR 20 are not exceeded.

Specification:

4.29.1 The quantity of radioactive material contained in an outside temporary radioactive liquid storage tank shall be determined to be within the limit of Specification 3.25.1 by analyzing a representative sample of the contents of the tank at least once per 7 days when radioactive materials are being added to the tank.

Bases:

This specification is provided to ensure that in the event of an uncontrolled release of the contents of the tank the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in the unrestricted area.

A2

5.5.12.

4.29.2 ~~Radioactive Gas Storage Tanks~~

~~Applicability: At all times~~

~~Objective: To ensure meeting the requirements of Specification 3.25.2.~~

~~Specification:~~

~~4.29.2 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limits of Specification 3.25.2 at least once per 24 hours when radioactive materials are being added to the tank and the reactor coolant activity exceeds the limits of Specification 3.1.4.1.b.~~

LAS
EGO STRMP

~~Bases:~~

~~This specification is provided so that the requirements of Specification 3.25.2 are met.~~

A2

4.1

5.0 DESIGN FEATURES

Specifications for design features are intended to cover characteristics of importance to each of the physical barriers, and to the maintenance of safety margins in the design.

A1

5.1 SITE

Applicability

Applies to the location and extent of the exclusion area.

Objective

To define the location and the size of the site area as pertains to safety.

Specification

4.1 Arkansas Nuclear One-Unit 1 is located on a site consisting of approximately 1100 acres which provides for 0.65 statute mile exclusion radius from the reactor building. This exclusion area includes certain portions of the bed and banks of the Dardanelle Reservoir which are owned by the Federal Government. An easement authorizes exclusion of all persons from these areas during periods of emergency. The site is approximately 6 statute miles WNW from the City of Russellville (Latitude 35°-18'-36" N, Longitude 93°-13'-53" W) in an area characterized by remoteness from population centers.

LA1

SAR

REFERENCE

FSAR, Section 2.2

4.0

LA1

SAR

5.2 REACTOR BUILDING

Applicability

Applies to those design features of the reactor building relating to operational and public safety.

Objective

To define the significant design features of the reactor building structure, reactor building isolation system, and penetration room ventilation system.

Specification

5.2.1 Reactor Building Structure

The reactor building completely encloses the reactor and the associated reactor coolant system. It is a fully continuous reinforced concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post tensioning system. The foundation slab is conventionally reinforced with high strength reinforcing steel. The entire structure is lined with 1/4" welded steel plate to provide vapor tightness.

The internal net free volume of the reactor building is approximately 1.81×10^6 cu. ft. The approximate inside dimensions are: diameter 116'; height--207'. The approximate thickness of the concrete forming the buildings are: cylindrical wall--3-3/4'; dome--3-1/4'; and the foundation slab--9'.

The concrete reactor building structure provides adequate shielding for both normal operation and accident situations. Design pressure and temperature are 59 psig and 286 F, respectively.

The reactor building is designed for an external atmospheric pressure of 3.0 psi greater than the internal pressure. This corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 110 F and it is subsequently cooled to an internal temperature of less than 50 F. Since the building is designed for this pressure differential, vacuum breakers are not required.

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in FSAR Section 14 with no loss of integrity. In this event the total energy contained in the water of the reactor coolant system is

3.7.11

<LATER>
(4.0)

assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safety features, and the combined influence of energy sources and heat sinks. [1]

LATER

5.2.2 Reactor Building Isolation System

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves. [2]

5.2.3 Penetration Room Ventilation System

This system is designed to collect, control, and minimize the release of radioactive material from the reactor building to the environment in post-accident conditions. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms. When the system is in operation, a slightly negative pressure will be maintained in the penetration room to assure inleakage. [3]

LA1
Bases

<LATER>
(4.0)

REFERENCES

(1) FSAR Section 5.1

(2) FSAR Section 5.2.5

(3) FSAR Section 6.5

LATER

LA1
Bases

4.0

LA1

SAR

assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safety features, and the combined influence of energy sources and heat sinks. ([1])

5.2.2 Reactor Building Isolation System

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves. ([2])

5.2.3 Penetration Room Ventilation System

LATER

⟨LATER⟩ (3.7)

This system is designed to collect, control, and minimize the release of radioactive material from the reactor building to the environment in post-accident conditions. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms. When the system is in operation, a slightly negative pressure will be maintained in the penetration room to assure inleakage. ([3])

REFERENCES:

(1) FSAR Section 5.1

(2) FSAR Section 5.2.5

(3) FSAR Section 6.5

LA1

SAR

⟨LATER⟩ (3.7)

LATER

4.2

5.3 REACTOR

Specification

5.3.1 Reactor Core

4.2.1

5.3.1.1 The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide pellets. Limited substitutions of stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. A1

(UO_2) as fuel material.

5.3.1.2 The reactor core approximates a right circular cylinder with an equivalent diameter of 128.9 inches and an active height of 144 inches. The active fuel length is approximately 142 inches.[2] LA1 SAR

5.3.1.3 The average enrichment of the initial core is a nominal 2.62 weight percent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is less than 3.5 weight percent U-235. LA1 SAR

4.2.2

5.3.1.4 There are 60 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSRA) distributed in the reactor core as shown in FSAR Figure 3-60. The full-length CRA contain a 134-inch length of silver-indium-cadmium alloy, and the clad with stainless steel. Each APSRA contains a 63-inch length of Inconel-600 alloy[3], as approved by the NRC. LA1 SAR A1

control material of

5.3.1.5 The initial core had 68 burnable poison spider assemblies with similar dimensions as the full-length control rods. The cladding is Zircaloy-4 filled with alumina-boron and placed in the core as shown in FSAR Figure 3-2. LA1 SAR

5.3.1.6 Reload fuel shall conform to the design and evaluation described in FSAR and shall not exceed an enrichment of 4.1 weight percent of U-235. LA1 SAR

5.3.2 Reactor Coolant System

5.3.2.1 The reactor coolant system is designed and constructed in accordance with code requirements.[4] LA1 SAR

4.2

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, are designed for a pressure of 2500 psig and a temperature of 650 F. The pressurizer and pressurizer surge line are designed for a temperature of 670 F.(5)

LA1
SAR

5.3.2.3 The reactor coolant system volume is less than 12,200 cubic feet.

LA1
TRM

REFERENCES:

(1) FSAR, Section 3.2.1

(2) FSAR, Section 3.2.2

(3) FSAR, Section 3.2.4.2

(4) FSAR, Section 4.1.3

(5) FSAR, Section 4.1.2

A1

4.3

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

A1

~~Applicability~~

~~Applies to storage facilities for new and spent fuel assemblies.~~

~~Objective~~

~~To assure that both new and spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.~~

Specification

5.4.1 New Fuel Storage

4.3.1.2.a-e

LA1 TRM

1. New fuel assemblies may be stored in the Fresh Fuel Storage Rack (FFSR). The FFSR consists of ~~a nine by eight array of~~ storage cells on nominal center to center distance of 21 inches in both directions. Ten interior storage cells, as shown in Figure 5.~~4-1~~, are precluded from use ~~and will be physically blocked prior to any storage in the fresh fuel rack~~. This configuration is sufficient to maintain a K_{eff} of less than 0.98 with optimum moderation and 0.95 under normal conditions, based on fuel with an enrichment of 4.1 weight percent U-235.

LA1 TRM

A3

2. New fuel may also be stored in the spent fuel pool ~~or in its shipping containers~~.

LA1 SAR

4.3.1.1.d
4.3.1.1.e

5.4.2 Spent Fuel Storage

4.3.1.1.b

1. The spent fuel racks are designed and shall be maintained so that the calculated effective multiplication factor is no greater than 0.95 (including all known uncertainties) when the pool is flooded with unborated water.

~~2. The spent fuel pool and the new fuel pool racks are designed as seismic Class I equipment.~~

LA1 SAR

~~REFERENCES~~

~~FSAR, Section 9.6~~

A1

<Add 4.3.1.1.c>

M1

4.3

Fig. 4.3.1.2-1 ~~FIGURE 5.4-1~~ ANO FFSR LOADING PATTERN A1

<-----NORTH

		NO			NO		
			NO	NO			
			NO	NO			
			NO	NO			
		NO			NO		

"NO" Indicates a location in which fuel loading is prohibited.

5.1
5.2.1
5.2.2

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

5.1.1 ~~6.1.1~~ The ANO-1 plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

5.1.2 ~~6.1.2~~ An individual with an active Senior Reactor Operator (SRO) license shall be designated as responsible for the control room command function while the unit is above the Cold Shutdown condition. With the unit not above the Cold Shutdown condition, an individual with an active SRO license or Reactor Operator license shall be designated as responsible for the control room command function.

MODES 1, 2, 3, & 4

A1

6.2 ORGANIZATION

5.2.1 ~~6.2.1~~ OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the unit specific titles of those personnel fulfilling its responsibilities of the positions delineated in these Technical Specifications shall be documented in the Safety Analysis Report (SAR).

b. The ANO-1 plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate executive shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety. The specified corporate executive shall be documented in the SAR.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

UNIT STAFF

5.2.2.f ~~6.2.2~~ The operations manager or the assistant operations manager shall hold a senior reactor operator license. ~~Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.~~

A5

5.2.2
5.3.1

5.2.2.e ~~6.2.2.1~~ Administrative controls shall be established to limit the amount of overtime worked by plant staff performing safety-related functions. These administrative controls shall be in accordance with the guidance provided by the NRC Policy Statement on working hours (Generic Letter 82-12).

6.3. FACILITY STAFF QUALIFICATIONS

ANS 3.1-1978

5.3.1 ~~6.3.1~~ Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable position, except for the designated radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. M1

~~6.4 DELETED~~

~~6.5 DELETED~~

A1

5.2.2

Table 6.2-1

ARKANSAS NUCLEAR ONE

MINIMUM SHIFT CREW COMPOSITION #

UNIT 1

A1

	LICENSE CATEGORY	ABOVE COLD SHUTDOWN	COLD AND REFUELING SHUTDOWNS	
5.2.2.b	SOL	2	1*	A5
	OL	2	1	
5.2.2.a	NON-LICENSED	~~3~~ 2	1	LA2 SAR

~~*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising refueling operations after the initial fuel loading.~~ A5

5.2.2.c #Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements ~~of Table 6.2-1~~. A1

Additional Requirements:

~~1. At least one licensed Operator shall be in the control room when fuel is in the reactor.~~

~~2. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.~~ A5

5.2.2.d ~~3.~~ An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

~~4. All refueling operations after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.~~ A5

5.2.2.g 5. ~~When the unit is above the Cold Shutdown condition~~ In MODES 1, 2, 3, and 4 an individual shall provide advisory technical support for the unit operations shift supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. A1

6.6 DELETED

2,2 ~~6.7~~ ~~SAFETY LIMIT VIOLATION~~ SL Violations A1

~~6.7.1~~ The following actions shall be taken in the event a Safety Limit is violated:

2.2.1, 2.2.2, 2.2.3 ~~a.~~ The facility shall be placed in at least ~~hot shutdown~~ MODE 3 within one hour. M3

2.2.5 ~~b.~~ within 1 hour The Nuclear Regulatory Commission shall be notified pursuant to 10 CFR 50.72 ~~and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.6.~~ A1 A3

~~6.8 PROCEDURES AND PROGRAMS~~

~~6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:~~

~~a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.~~

~~b. Refueling operations.~~

~~c. Surveillance and test activities of safety related equipment.~~

~~d. (Deleted)~~

~~e. (Deleted)~~

~~f. Fire Protection Program Implementation.~~

~~g. New and spent fuel storage.~~

~~h. Offsite Dose Calculation Manual and Process Control Program implementation at the site.~~

<LATER> (5.0) LATER

2.2.4 In MODES 3, 4 and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes M2

LAR A4

5.4.1
5.5.1
5.5.3

6.6 ~~DELETED~~ (A1)

<LATER> (2.0)

6.7 ~~SAFETY LIMIT VIOLATION~~

6.7.1 ~~The following actions shall be taken in the event a Safety Limit is violated:~~ LATER

a. ~~The facility shall be placed in at least hot shutdown within one hour.~~

b. ~~The Nuclear Regulatory Commission shall be notified pursuant to 10 CFR 50.72 and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.6.~~

6.8 PROCEDURES AND PROGRAMS

5.4.1 6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

5.4.1.a a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, ~~November, 1972~~. Rev 2, Feb 1978 (M3)

b. ~~Refueling operations.~~

c. ~~Surveillance and test activities of safety related equipment.~~ (A1)

d. ~~(Deleted)~~

e. ~~(Deleted)~~

5.4.1.d f. Fire Protection Program Implementation.

g. ~~New and spent fuel storage.~~ (A1)

5.4.1.c (5.5.1) h. Offsite Dose Calculation Manual ~~and Process Control Program~~ implementation at the site. (A1)

<ADD: 5.4.1.c for "other programs">
<ADD: 5.4.1.b for EOPs per GL 82-33> (M3)

 (A15)

LAR

5.4.1

5.5.16

~~6.8.2 (Deleted)~~

~~6.8.3 (Deleted)~~ A1

5.5.16 ~~6.8.4~~ The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained: A1

A program shall be established to implement the leakage rate testing of the reactor building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated reactor building internal pressure for the design basis loss of coolant accident, P_a, is 54 psig.

The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of ~~containment~~ Reactor building air weight per day at P_a. A1

Reactor building leakage rate acceptance criteria is $\leq$ 1.0 L_a. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are ~~$\leq$~~ < 0.60 L_a for the Type B and Type C tests and ~~$\leq$~~ < 0.75 L_a for Type A tests. A11

The provisions of ~~Specification 4.0.2~~ SR 3.0.2 do not apply to the test frequencies specified in the Reactor Building Leakage Rate Testing Program. A1

The provisions of ~~Specification 4.0.3~~ SR 3.0.3 are applicable to the Reactor Building Leakage Rate Testing Program. A1

5.5.4 6.8.5 **The Radioactive Effluent Controls Program shall be established, implemented, and maintained:**

This program conforms with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. **Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;**

b. **Limitations on the concentrations of radioactive material released ~~in liquid effluents to unrestricted~~ areas conforming to ~~10 CFR 20, Appendix B, Table II, Column 2;~~** A17

<INSERT 127a A>

c. **Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;**

d. **Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;**

e. **Determination of cumulative ~~and projected~~ dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM ~~at least every 31 days;~~** A17

<INSERT 127a B>

f. **Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;**

g. **Limitations on the dose rate resulting from radioactive material released in gaseous effluents** from the site **to areas** at or **beyond the site boundary ~~conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;~~** A1 A17

<INSERT 127a C>

h. **Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;**

i. **Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and**

j. **Limitations on the annual dose or dose commitment to any member of the public**, beyond the site boundary, **due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.** A1

<INSERT 127a D> A17

<CTS INSERT 127aA>

... ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402.

<CTS INSERT 127aB>

. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM ...

<CTS INSERT 127aC>

... shall be in accordance with the following:

1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

<CTS INSERT 127aD>

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.0

6.9 DELETED (A1)

5.7

A5

~~6.10 RADIATION PROTECTION PROGRAM~~

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

5.7 **6.11 HIGH RADIATION AREA**

5.7.1 **6.11.1 In lieu of the "control device" or "alarm signal" required by ~~paragraph 20.203(c)(2) of~~ 10CFR20, each high radiation area (as defined in ~~20.202(b)(3) of 10CFR20~~) in which the intensity of radiation is 1000 ~~mrem/hr or less~~ shall be barricaded and conspicuously posted as a high radiation area and shall be controlled by requiring the issuance of a radiation work permit. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:** A1 L2

5.7.1.a

5.7.1.b

<INSERT 129 A>

<INSERT 129 B>

5.7.1.d

5.7.1.d.1 ~~a.~~ **A radiation monitoring device which continuously indicates the radiation dose rate in the area.**

5.7.1.d.2 ~~b.~~ **A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a pre-set integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.** L2 A1

only

<INSERT 129 C>

5.7.1.e

<INSERT 129 D>

5.7.1.d.4.i ~~c.~~ **An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation work permit.**

<Note: this text is repeated in insert 129C>

<CTS INSERT 129aA>

... at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

   1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

   2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

   3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

      (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

      (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with ~~and control every~~ individuals in the area. S

   4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

who are covered by such surveillance A1

5.7

5.7.2 ~~6.11.2~~ ~~The requirements of 6.11.1 above, shall also apply to each~~ high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. ~~In addition, locked doors shall be provided to prevent unauthorized entry into such areas and access to these areas shall be maintained under the administrative control of the shift supervisor on duty and/or the designated radiation protection manager.~~ L2

<INSERT 129a A>

<CTS INSERT 129aA>

... at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

   1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

   2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

   3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

      (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

      (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

   4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

<CTS INSERT 129aA>
(continued)

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. . These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

5.6
5.6.1

6.12 REPORTING REQUIREMENTS

6.12.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Administrator of the appropriate NRC Regional Office unless otherwise noted. A5

6.12.2 Routine Reports

6.12.2.1 Startup Report

A summary report of plant startup and power escalation testing shall be submitted following 1) receipt of an operating license, 2) amendment to the license involving a planned increase in power level, 3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and 4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. LA6 TRM

Startup reports shall be submitted within 1) 90 days following completion of the startup test program, 2) 90 days following resumption or commencement of commercial power operation, or 3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

5.6.1 6.12.2.2 Occupational Exposure Data Report 1/ L3a

An Occupational Exposure Data Report for the previous calendar year shall be submitted by April 30 ~~prior to March 1~~ of each year. The report shall contain a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) , for whom monitoring was performed, receiving an annual deep dose equivalent ~~exposures~~ greater than 100 mrem/~~yr~~ and their associated collective deep dose equivalent (reported in person- ~~man~~ rem) ~~exposure~~ according to work and job functions, 2/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. A13

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2/ This tabulation supplements the requirements of ~~20.407 of~~ 10 CFR ~~Part~~ 20.2206.

5.6.1 Note....

5.6.1
5.6.2
5.6.3

ionization chamber — electronic dosimeter

5.6.1 The dose assignments to various duty functions may be estimates based on pocket ~~dosimeter~~ TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total ~~whole body dose~~ deep dose equivalent received from external sources shall be assigned to specific major work functions. A13

5.6.4 6.12.2.3 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis by the 15th of each month following the calendar month covered by the report.

~~6.12.2.4 Annual Report~~

~~All challenges to the pressurizer electromatic relief valve (ERV) and pressurizer safety valves shall be reported annually.~~ L3b

5.6.2 6.12.2.5 Annual Radiological Environmental Operating Report *

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 6.12.2.6 Radioactive Effluent Release Report **

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year ~~in accordance with 10 CFR 50.36a~~. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1. A14

5.6.2 NOTE * A single submittal may be made for ANO. The submittal should combine those sections that are common to both units.

5.6.3 NOTE ** A single submittal may be made for ANO. The submittal ~~should~~ shall combine those sections that are common to both units. The submittal shall specify the releases of radioactive material from each unit.

5.6.5

5.6.5

6.12.3 CORE OPERATING LIMITS REPORT

6.12.3.1 The core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle for the following Specifications:

- 2.1 Safety Limits, Reactor Core - Axial Power Imbalance protective limits and Variable Low RCS Pressure-Temperature Protective Limits
- 2.3.1 Reactor Protection System trip setting limits - Protection System Maximum Allowable Setpoints for Axial Power Imbalance and Variable low RC system pressure
- 3.1.8.3 Minimum Shutdown Margin for Low Power Physics Testing
- 3.5.2.1 Allowable Shutdown Margin limit during Power Operation
- 3.5.2.2 Allowable Shutdown Margin limit during Power Operation with inoperable control rods
- 3.5.2.4 Quadrant power Tilt limit
- 3.5.2.5 Control Rod and APSR position limits
- 3.5.2.6 Reactor Power Imbalance limits

6.12.3.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specification shall be those previously reviewed and approved by the NRC in Babcock & Wilcox Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.

6.12.3.3 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.12.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC ~~Document Control Desk with copies to the Regional Administrator and Resident Inspector.~~ A5

Note: Next non-blank page is 146

5.6.6

5.6.6 ~~6.12.4~~ Reactor Building Inspection Report

~~6.12.4.1~~ Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The reportshall include the cause of the condition that does not meet the acceptance criteria, the applicability of the conditions to the other unit, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion date of necessary repairs, and the extent, nature, and frequency of additional examinations.

6.12.5 Special Reports

<LATER> (5.0) Special reports shall be submitted to the Administrator of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification. LATER

a. Deleted A1

<LATER> (3.3D) b. Inoperable Containment Radiation Monitors, Specification 3.5.1, Table 3.5.1-1. LATER

c. Deleted A1

<LATER> (5.0) d. Steam Generator Tubing Surveillance - Category C-3 Results, Specification 4.18. LATER

e. Miscellaneous Radioactive Materials Source Leakage Tests, Specification 3.12.2. L11

f. Deleted

g. Deleted

h. Deleted

i. Deleted A1

<LATER> (3.8) j. Degraded Auxiliary Electrical Systems, Specification 3.7.2.K. LATER

<LATER> (3.3D)

k. Inoperable Reactor Vessel Level Monitoring Systems, Table 3.5.1-1

l. Inoperable Hot Leg Level Measurement Systems, Table 3.5.1-1

m. Inoperable Main Steam Line Radiation Monitors, Specification 3.5.1, Table 3.5.1-1. LATER

6.12.5 **Special Reports**

<LATER> (5.0) Special reports shall be submitted to the Administrator of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification. LATER

a. Deleted A1

b. Inoperable Containment Radiation Monitors, Specification 3.5.1, Table 3.5.1-1. LA1 Bases

c. Deleted A1

<LATER> (5.0) d. Steam Generator Tubing Surveillance - Category C-3 Results, Specification 4.18. LATER

<LATER> (3.7) e. Miscellaneous Radioactive Materials Source Leakage Tests, Specification 3.12.2. LATER

f. Deleted

g. Deleted A1

h. Deleted

i. Deleted

<LATER> (3.8) j. Degraded Auxiliary Electrical Systems, Specification 3.7.2.H. LATER

k. Inoperable Reactor Vessel Level Monitoring Systems, Table 3.5.1-1 LA1 Bases

l. Inoperable Hot Leg Level Measurement Systems, Table 3.5.1-1

m. Inoperable Main Steam Line Radiation Monitors, Specification 3.5.1, Table 3.5.1-1. L14

3.8

6.12.5 Special Reports

<LATER> (5.0) Special reports shall be submitted to the Administrator of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification. — LATER

a. Deleted

<LATER> (3.3 D) b. Inoperable Containment Radiation Monitors, Specification 3.5.1, Table 3.5.1-1. — LATER

c. Deleted

<LATER> (5.0) d. Steam Generator Tubing Surveillance - Category C-3 Results, Specification 4.18. — LATER

<LATER> (3.7) e. Miscellaneous Radioactive Materials Source Leakage Tests, ~~Specification 3.12.2.~~ — LATER

f. Deleted

g. Deleted

h. Deleted

i. Deleted

j. Degraded Auxiliary Electrical Systems, Specification 3.7.2.H. — M8

~~k. Inoperable Reactor Vessel Level Monitoring Systems, Table 3.5.1-1~~

<LATER> (3.3 D) l. Inoperable Hot Leg Level Measurement Systems, Table 3.5.1-1 — LATER

m. Inoperable Main Steam Line Radiation Monitors, Specification 3.5.1, Table 3.5.1-1.

5.6.7

5.6 6.12.5 Special Reports

Special reports shall be submitted to the Administrator of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification. A5

a. Deleted A1

<LATER> (3.3D) b. Inoperable Containment Radiation Monitors, Specification 3.5.1, Table 3.5.1-1. LATER

c. Deleted A1

5.6.7 d. Steam Generator Tubing Surveillance - Category C-3 Results, Specification ~~4.18~~ 5.5.9

<LATER> (3.7) e. Miscellaneous Radioactive Materials Source Leakage Tests, Specification 3.12.2. LATER

f. Deleted

g. Deleted

h. Deleted

i. Deleted A1

<LATER> (3.8) j. Degraded Auxiliary Electrical Systems, Specification 3.7.2.M. LATER

k. Inoperable Reactor Vessel Level Monitoring Systems, Table 3.5.1-1

<LATER> (3.3D) l. Inoperable Hot Leg Level Measurement Systems, Table 3.5.1-1 LATER

m. Inoperable Main Steam Line Radiation Monitors, Specification 3.5.1, Table 3.5.1-1.

5.5.14
5.5.15

~~6.13 (DELETED)~~ A1

Add ITS 5.5.14, "Tech Spec Bases Control Program" M7

Add ITS 5.5.15, "Safety Function Determination Program M7

5.5.1

5.5.1

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports ~~required by Specifications 6.12.2.5 and 6.12.2.6.~~ A1

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after approval of the General Manager, ~~Plant Operations~~; AND and A1 SAR

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed and shall also indicate the date (i.e., month and year) the change was implemented.