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ATTACHMENT 1

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to add new Radiological Effluent Technical Specifications necessary to implement the requirements of 10 CFR 50 Appendix I. The amendment provides revised technical specifications for Section 3/4.11, "Radioactive Effluents", Section 3/4.12, "Radiological Environmental Monitoring", and several related specifications and their Bases.

1. Present condition of license:

As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Section 1.0, "Definitions" (pages 1-3, 1-4, 1-5, 1-6 and 1-7), Section 3/4.3, "Instrumentation" (pages 3-59 through 3-68), Section 3/4.11, "Radioactive Effluents" (pages 11-1 through 11-19), Section 3/4.12, "Radiological Environmental Monitoring" (pages 12-1 through 12-10), Section B3/4, "Bases", (pages B3/4 3-4 and B3/4 11-1 through B3/4 12-2), and Section 5, "Design Features" (page 5-1), and Section 6, "Administrative Controls" (pages 6-16 to 6-18).

2. Proposed condition of license:

Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to add new definitions for Definition 1.19, "Member(s) of the Public" (page 1-4), Definition 1.32, "Site Boundary" (page 1-6), and Definition 1.40, "Unrestricted Area" (page 1-7); to revise definitions for Definition 1.25, "Process Control Program" (page 1-5), Definition 1.35, "Source Check" (page 1-6) and Definition 1.34, "Solidification" (page 1-6). Add minor changes to the Technical Specifications for Definition 1.11, "Dose Equivalent I-131" (page 1-3), Specification 3/4.3.3.9, "Radioactive Effluent Monitoring Instrumentation"(pages 3-59 through 3-63), Specification 3/4.3.3.10, "Radioactive Gaseous Effluent Monitoring Instrumentation" (pages 3-64 through 3-68), Section 5 "Design Features" (page 5-1) and Section 6, "Administrative Controls" (pages 6-16 to 6-18). Revise extensively Section 3/4.11, "Radioactive Effluents" (pages 11-1 through 11-19), and Section 3/4.12, "Radiological Environmental Monitoring" (pages 12-1 through 12-10); and to revise the applicable areas of Section B3/4, "Bases" for these specifications (pages B3/4 3-4 and B3/4 11-1 through B3/4 12-2).

3. Justification:

The changes described are to update the Radiological Effluent Technical Specifications to be consistent with draft Revision 5 of the Westinghouse Standard Technical Specifications and current NRC staff positions.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (ii) not likely to involve a significant hazards consideration is a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications; for example, a more stringent surveillance requirement. The proposed changes fit this example in that additional restrictions will be imposed with the addition of the new Radiological Effluent Technical Specifications.

ATTACHMENT 2

REPORTING REQUIREMENTS

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to bring it into conformance with the reporting requirements of 10 CFR 50.73. The amendment changes the Technical Specifications in accordance with the guidance provided in Generic Letter 83-43, "Reporting Requirements of 10 CFR Part 50, Sections 50.72 and 50.73, and Standard Technical Specifications", for the Licensee Event Report System.

The amendment also revises the action statements of the Limiting Condition for Operation of Specification 3.3.3.8, "Fire Detection Instrumentation", Specification 3.7.9.1, "Fire Suppression Water System", Specification 3.7.9.2, "Spray and/or Sprinkler Systems", Specification 3.7.9.3, "CO_2 Systems", Specification 3.7.9.4, "Halon System", Specification 3.7.9.5, "Fire Hose Stations", and Specification 3.7.10, "Fire Barrier "Penetrations Penetrations", to remove the requirement for making a special report. The revision of these specifications is in accordance with guidance provided by the staff during discussions with PGandE.

1. Present condition of license:

   As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Section 1.0, "Definitions" (page 1-6), Section 3/4.3, "Instrumentation" (page 3-55), Section 3/4.4, "Reactor Coolant System" (pages 4-15, 4-17, 4-25, and 4-26), Section 3/4.6, "Containment Systems" (page 6-9), Section 3/4.7, "Plant Systems" (pages 7-26, 7-28, 7-30, 7-32, 7-33, and 7-36), Section 3/4.8, "Electrical Power Systems" (page 8-7), Section B3/4, "Bases" (pages B3/4 4-3 and B3/4 7-7), and Section 6.0, "Administrative Controls" (pages 6-7, 6-10, 6-12, 6-19, 6-20, 6-21, and 6-22).

2. Proposed condition of license:

   Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to revise: Definition 1.30, "Reportable Event" (page 1-6), Surveillance Requirement 4.4.5.5c of the "Steam Generators" specification (pages 4-15 and 4-17); Action Statement a. (Modes 1, 2, 3, 4 and 5) of Limiting Condition for Operation 3.4.8 of the "Specific Activity" specification (pages 4-25 and 4-26); Surveillance Requirement 4.6.1.6.2 of the "Containment Structural Integrity" specification (page 6-9), Surveillance Requirement 4.8.1.1.4 of the "A.C. Sources" specification (page 8-7), the Bases for Specification 3/4.4.5, "Steam Generators" (page B3/4 4-3), and Sections 6.5.1.6f, 6.5.2.7g, 6.6, 6.9 and 6.10.1c of the "Administrative Controls" section (pages 6-8, 6-11, 6-12, 6-13, and 6-20) to include the new reporting requirements of 10 CFR 50.73.

Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to revise: Action Statement a.2 of Specification 3.3.3.8, "Fire Detection Instrumentation" (page 3-54); Action Statement b.2 of Specification 3.7.9.1, "Fire Suppression Water System" (page 7-26); Action Statement a. of Specification 3.7.9.2, "Spray and/or Sprinkler System" (page 7-28); Action Statement a. of Specification 3.7.9.3, "CO_2 System" (page 7-30); Action Statement a. of Specification 3.7.9.4, "Halon System" (page 7-32); Action Statement a. of Specification 3.7.9.5, "Fire Hose Stations" (page 7-33); Action Statement a. of Specification 3.7.10, "Fire Barrier Penetrations" (page 7-36), and the Bases for Specification 3/4.7.9, "Fire Suppression Systems" (page B3/4 7-6) to remove the requirement for making a special report.

3. Justification:

The changes described are to (1) revise the Technical Specifications to reflect the new reporting requirements of 10 CFR 50.73, in accordance with the guidance provided in Generic Letter 83-43, "Reporting Requirements of 10 CFR Part 50, Sections 50.72 and 50.73, and Standard Technical Specifications", for the Licensee Event Report system and (2) revise the action statements of those fire protection specifications that require a special report to remove this requirement, in accordance with guidance provided by the NRC Staff during discussions on technical specification revision.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (vii) of actions not likely to involve a significant hazards consideration is a change to make a license conform to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations. The proposed changes, to incorporate the new reporting Requirements of 10 CFR 50.73, fit this example in that the changes reflect guidance provided by the NRC in new regulations.

B. (Continued)

Another example is, (vi), a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method. The proposed change to remove the requirement for making a special report, in accordance with guidance provided by the NRC Staff during discussions on technical specifications, is similar to the.example in that the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. However, this change is more conservative than the example, in that the probability and consequences of previously-analyzed accidents are not increased and safety margins are not reduced.

ATTACHMENT 3

ADMINISTRATIVE CHANGES

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to eliminate typographical errors, provide additional clarification, improve consistency and adjust nomenclature to improve format and legibility. These proposed additions and revisions do not change the technical content of the specifications. Examples of some of these changes are: (a) typographical, to correct all the symbol "l"s that were inadvertently used for the number "1" in the original typing, (b) clarification, to the Bases Section 2.1.1 to reflect the present condition of the License by revising the DNB correlation from "W-3" to "R-Grid", (c) consistency, Action Statement 3.5.1b was revised from ".. be in Hot Standby within one hour and be in Hot Shutdown within the next 12 hours" to ".. be in Hot Standby within 6 hours and be in Hot Shutdown within the next 6 hours" to be consistent with the statement of Specification 3.0.3., and (d) nomenclature, in a footnote for Specification 3.4.1.4.1 the word "Loop" was revised to "Train" to correctly describe that portion of the RHR system. The revisions required to incorporate the changes in the areas listed were identified by the NRC Staff and PGandE during discussions on Technical Specification revision.

1. Present condition of license:

   As described on the applicable pages of the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), all sections.

2. Proposed condition of license:

   Administrative changes as described in the marked-up copy of the Technical Specification (Attachment 22) that include:

   a. <u>Typographical Corrections</u> - Typographical corrections have been made throughout the Technical Specifications. These changes do not affect the meaning or intent of the specifications and include spelling, verb person, and punctuation corrections. Also, the symbol used for the numeral one shall be changed from "l" to "1", where appropriate during retyping.

   b. <u>Clarifications</u> - Language clarifications have been made in various specifications. These clarifications in no way change the substance or intent of the specifications and are wholly consistent with the underlying safety analyses and evaluations. Examples of clarification changes are shown in Bases 2.1.1 (Page B 2-1) to revise the referenced DNB correlation from "W-3" to reflect the current analysis and in Surveillance Requirement 4.10.3.2 (Page 10-3), the test was changed from a "Channel Functional Test" to "An Analog Channel Operational Test," to reflect the correct test to be performed.

A.2.b.(Continued)

Clarification changes have been made in Specifications 2.1.1 (page 2-1) and 2.1.2 (page 2-1); Bases 2.1.1 (page B 2-1), 2.1.2 (page B2-2) and 2.2.1 (page B 2-4); Specification 3.0.4 (page 0-1), Specification 4.0.6 (page 0-5), Specification 3/4.1.1.3 (page 1-4), Specification 3/4.1.3.1 (pages 1-15, 1-16), Specification 3/4.1.3.3 (page 1-19); Tables 3.2-1 (page 2-18), 4.3-1 (page 3-13) and 3.3-3 (pages 3-15 through 3-18, and 3-22); Table 3.3-4 (page 3-23), Table 3.3-5 (page 3-28); Specification 3/4.3.3.5 (page 3-48); Table 3.3-9 (page 3-49); Table 4.3-6 (page 3-50), Specification 3/4.3.3.9 (page 3-59); Tables 3.3-12 (page 3-61), Table 3.3-13 (page 3-65), and Table 3.3-13 (page 3-66); Table 4.3-9 (pages 3-67 and 3-68), Specifications 3/4.3.4.1 (page 3-69), 3/4.4.1.3 (page 4-3 and 4-4), 3/4.4.1.4.1 (page 4-5), 3/4.4.1.4.2 (page 4-6), 3/4.4.2.1 (page 4-7), 3/4.4.4 (page 4-10), 3/4.4.6.1 (page 4-18) and 3/4.4.8 (page 4-25); Table 4.4-4 (page 4-27); Figures 3.4-2 and 3.4-3 (pages 4-30 and 4-31), Specification 3/4.5.3 (page 5-8), Specifications 3/4.5.4.1 (page 5-9), 3/4.5.5 (page 5-11), 3/4.6.1.3 (page 6-6); Specification 3/4.6.3 (page 6-15), Table 3.7-1 (page 7-2); Specifications 3/4.7.1.5 (page 7-9), 3/4.7.9.1 (page 7-27), Specification 3/4.7.9.2 (page 7-28), 3/4.7.11 (page 7-37),Specification 3/4.8.1.1 (pages 8-1, 8-3 and 8-6), Specification 3/4.8.1.2 (page 8-11), Specification 3/4.8.2.1 pages 8-12 and 12a), Specification 3/4.8.2.2 (page 8-13), Specification 3/4.8.3.1 (pages 8-15 and 8-16), Specification 3/4.8.4.1 (page 8-18), 3/4.9.1 (page 9-1), 3/4.9.2 (page 9-2), 3/4.9.6 (page 9-6), 3/4.9.8.2 (page 9-9), 3/4.9.10 (page 9-11), 3/4.10.3 (page 10-3) and 3/4.10.5 (page 10-5); and Bases 3/4.0 (page B0-1 and B0-2), 3/4.1.1.4 (page B1-2), 3/4.3.1 and 3/4.3.2 (page B3-1), 3/4.3.3.6 (page B3-3), 3/4.4.2 (page B4-2), 3/4.6.2 (page B4-4), 3/4.4.8 (page B4-5 and B4-6), 3/4.4.9 (page B4-12, B4-13, and B4-16), 3/4.7.1.4 (page B7-2), Bases 3/4.8.1, 3/4.8.2, 3/4.8.3 (pages B8-1 and B8-2) and 3/4.9.1.3 (page B9-3).

c. <u>Consistency Changes</u> - Changes have been made, where appropriate, to improve uniformity within and between specifications that describe, reference, or affect the same or related systems. Examples of consistency changes are shown in Definition 1.12 (page 1-3), where the qualifier of 15-minute half-lives has been removed for consistency with the Specific Activity Specifications; in 3.9.13 (page 9-16), where reference to the FSAR has been relocated to the Bases section; and in Statement 3.5.1b (page 5-1) where the shutdown times have been revised from "...be in Hot Standby within one hour and be in Hot Shutdown within the next 12 hours" to "...be in Hot Standby within 6 hours and be in Hot Shutdown within the next 6 hours" to be consistent with Specification 3.0.3.

A.2.c. (Continued)

Consistency changes have been made in Definition 1.12 (page 1-3); Specification 3/4.2.1 (page 2-1 and 2-2); Table 3.3-1 (page 3-7); Table 3.3-12 (page 3-60), Specifications 3/4.4.1.1 (page 4-1), Specification 3/4.4.1.2 (page 4-2), Specification 3/4.4.9.1 (page 4-29), 3/4.5.1 (page 5-1) 3/4.6.1.1 (page 6-1), 3/4.6.2.1 (page 6-11), 3/4.6.3 (page 6-15), 7.1.1 (page 7-1), Specification 3/4.7.1.3 (page 7-6), 3/4.7.3.1 (page 7-11), Specification 3/4.7.5.1 (page 7-13), 3/4.8.1.1 (pages 8-1 and 8-2), Table 4.8-1 (page 8-8) and 3/4.9.13 (page 9-16); Bases 3/4.2.4 (page B2-6).

d. Nomenclature Changes - Nomenclature changes have been made, where appropriate, to the names used to designate systems, components, structures, or concepts. These changes do not affect the substance or intent of specifications. An example of a nomenclature change is use of an acronym to designate a system, i.e., changing "Engineering Safety Features" to "ESF".

Nomenclature changes have been made in Definition 1.8 (page 1-2), Definition 1.10 (page 1-2), Definition 1.16 (page 1-4) Definition 1.17 (page 1-4), Definition 1.35 (page 1-6); Bases 2.1.2 (page B2-2) and 2.2.1 (pages B2-3, B2-8 and B2-9); Specifications 3/4.1.1.1 (page 1-2), 3/4.1.2.2 (pages 1-8 and 1-9), 3/4.1.2.5 (page 1-12), 3/4.1.2.6 (page 1-13), 3/4.1.3.2 (page 1-18), 3/4.1.3.3 (page 1-19),and 3/4.2.1 (page 2-2); Table 4.3-3 (page 3-40), Table 4.3-8 (page 3-63), Table 4.3-9 (page 3-68), Specification 3/4.4.1.3 (pages 4-3 and 4-4), Specification 3/4.4.5.3 (page 4-13) and 3/4.4.8 (pages 4-25 and 4-26); Table 4.4-4 (page 4-27); Figure 3.4-1 (page 4-28), Specification 3/4.6.1.1 (page 6-1), 3/4.6.1.4 (page 6-7), 3/4.6.1.5 (page 6-8), 3/4.7.2.1 (page 7-10), 3/4.9.1 (page 9-1), 3/4.9.3 (page 9-3), 3/4.9.4 (page 9-4), 3/4.9.6 (page 9-6), 3/4.9.8.1 (page 9-8), 3/4.9.8.2 (page 9-9), 3/4.10.1 (page 10-1), and 3/4.10.5 (page 10-5); Bases 3/4.3.1 and 3/4.3.2 (page B3-2), Bases 3/4.4.1 (page B4-1) 3/4.4.4 (page B4-2), 3/4.4.5 (page B4-3), Bases 3/4.4.6.2 (page B4-4), Bases 3/4.4.8 (page B4-5), 3/4.6.1 (page B6-1), 3/4.7.1.4 (page B7-2), Bases 3/4.9.6 (page B9-2), 3/4.9.7 (page B9-2), and 3/4.9.8 (page B9-2); Specification 5.2.2 (page 5-1), Specification 6.5.1.6 (page 6-8), Specification 6.5.2.9 (page 6-12), Specification 6.8.4 (page 6-13), Specification 6.9.1 (page 6-15), Specification 6.9.1.4 (page 6-15), and Specification 6.9.1.7 (page 6-18).

3. Justification:

The changes described are to eliminate typographical errors, provide additional clarification, improve consistency, and adjust nomenclature to improve format and legibility of the Technical Specifications. These proposed additions and revisions were identified by the NRC Staff and PGandE during discussion on Technical Specification revisions.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendments involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendments would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (i) of actions not likely to involve a significant hazards consideration is a purely administrative change to the Technical Specifications. The proposed changes fit this example in that all of the changes are administrative in nature and do not change the technical content of the specifications.

ATTACHMENT 4

DELETED.

ATTACHMENT 5

ROD BOW PENALTY

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to delete the Rod Bow Penalty from Specification 3/4.2.3, "RCS Flowrate and Nuclear Enthalpy Rise Hot Channel Factor". This change is the result of the revised analysis described in the Westinghouse Rod Bow Topical Report (WCAP-8691, Rev. 1) and subsequently addressed in the related NRC SER.

1. Present condition of license:

   As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Specification 3/4.2.3, "RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor" (pages 2-9 through 2-13), that includes a factor for rod bow penalty in the calculation.

2. Proposed condition of license:

   Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to delete the Rod Bow Penalty from Specification 3/4.2.3, "RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor" (pages 2-9 through 2-12).

3. Justification:

   The NRC has reviewed the Westinghouse Rod Bow Topical Report (WCAP-8691, Rev. 1) and issued an SER on it. The Topical Report analysis and NRC SER are applicable to Diablo Canyon Units 1 and 2 and justify the deletion of the rod bow penalty.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (iv) not likely to involve a significant hazards consideration is a relief granted upon demonstration of acceptable operation from an operation restriction that was imposed because acceptable operation was not yet demonstrated. This assumes that the operating restriction and the criteria to be applied to a request for relief have been established in a prior review and that it is justified in a satisfactory way that the criteria have been met.

B.(Continued)

The proposed change fits this example in that the referenced Topical Report has been reviewed and approved in an issued NRC Safety Evaluation Report and is applicable to Diablo Canyon Units 1 and 2.

ATTACHMENT 6

STEAM LINE PRESSURE-LOW ALLOWABLE VALUE

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to change the allowable value for the Steam Line Pressure-Low signal to the Steam Flow High Safety Injection, listed in Table 3.3-4, "Engineered Safety Features Actuation System Instrumentation Trip Setpoint".

1. Present condition of license:

As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Table 3.3-4, "Engineered Safety Features Actuation System Instrumentation Trip Setpoints", Item 1.f.2, "Steam Line Pressure-Low" (page 3-23), and 4.d.2, "Steam Line Pressure-Low" (page 3-26), both have a setpoint of "≥ 585 psig" for their allowable value.

2. Proposed condition of license:

Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to revise the allowable value for the, "Steam Line Pressure-Low", setpoint in Table 3.3-4 (pages 3-23 and 3-26), to "≥ 580 psig".

3. Justification:

The change in allowable value for this setpoint is due to revised analysis provided by Westinghouse.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (iv) not likely to involve a significant hazards consideration is a relief granted upon demonstration of acceptable operation from an operation restriction that was imposed because acceptable operation was not yet demonstrated. This assumes that the operating restriction and the criteria to be applied to a request for relief have been established in a prior review and that it is justified in a satisfactory way that the criteria have been met.

B.(Continued)

The proposed change is similar to the example in that the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. However, this change is more conservative than the example, in that the probability and consequences of previously-analyzed accidents are not increased and safety margins are not reduced as demonstrated in the Westinghouse analysis.

ATTACHMENT 7

SEISMIC MONITORING INSTRUMENTATION

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to change Surveillance Requirement 4.3.3.3.2 to reflect new time intervals for the channel calibration and special report required following a seismic instrumentation actuation during a seismic event.

1. Present condition of license:

As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Surveillance Requirement 4.3.3.3.2 (page 3-42), that currently states that following the activation of the Seismic Monitoring Instrumentation (shown in Table 3.3-7) during a seismic event ".. a channel calibration, as applicable, [should be] performed within 5 days following the seismic event" and "a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days ..."

2. Proposed condition of license:

Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to revise Surveillance Requirement 4.3.3.3.2 (page 3-42) to say "... a channel calibration as applicable, [shall be] performed within 10 days following the seismic event" and "a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days ..."

3. Justification:

The proposed change allows for a more realistic time interval to perform a channel calibration on the Seismic Monitoring Instrumentation and to submit the subsequent special report. The increased time interval is due to the need to bring an outside vendor, to the plant, to perform the channel calibration on these monitors. This additional time was also reflected into the time interval for submitting the special report. The time intervals proposed are consistent with the draft Revision 5 of the Westinghouse Standard Technical Specifications and were supplied by the NRC Staff during discussion on Technical Specifications revisions and do not change the intent of the surveillance requirement.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different

B. (Continued)

kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (vi) of actions not likely to involve a significant hazards consideration is a change which may result in some increase to the probability or consequences of a previously-analyzed accident but that is clearly within all acceptable criteria in the Standard Review Plan.

The proposed change is similar to this example in that the surveillance interval to perform the test has been increased slightly to allow time for the vendor to perform the Channel Calibration. However, the change in time is minor and does not affect the acceptance criteria for the seismic monitoring system as described in the Standard Review Plan and is consistent with the draft Revision 5 of the Westinghouse Standard Technical Specifications.

ATTACHMENT 8

NOBLE GAS ACTIVITY MONITOR APPLICABILITY

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to add a notation to Table 3.3-13, "Radioactive Gaseous Effluent Monitoring Instrumentation"and Table 4.3-4, "Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements", to further define the applicability of the Noble Gas Activity Monitor associated with the Containment Purge System.

1. Present condition of license:

   As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Table 3.3-13, "Radioactive Gaseous Effluent Monitoring Instrumentation" and Table 4.3-9 "Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements," Instrument 5.a, "Noble Gas Activity Monitor - providing alarm and automatic termination of release (RM-14A and 14B)" (pages 3-65 and 3-66), presently has for its applicability "* at all times."

2. Proposed condition of license:

   Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) Table 3.3-13, "Radioactive Gaseous Effluent Monitoring Instrumentation" and Table 4.3-9 "Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements," (pages 3-64 to 3-68), to revise the applicability for Instrument 5.a, "Noble Gas Activity Monitor" to "*** Modes 1-4; also Mode 6 during CORE ALTERATIONS or movement of irradiated fuel within containment.

3. Justification:

   The proposed change only affects the "automatic termination" feature of the Noble Gas Activity Monitor (RM-14A or 14B), since the same monitor, with alarm, is required to be available for Instrument 3.a on Tables 3.3-13 and 4.3-9.

   When Containment Integrity is not required, in Mode 5 and a limited time in Mode 6, there is not need to provide automatic termination of release for the Containment Purge System.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

B. (Continued)

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (vi) of actions not likely to involve a significant hazards consideration is a change which may result in some increase to the probability or consequences of a previously-analyzed accident but that is clearly within all acceptable criteria in the Standard Review Plan.

The proposed change is similar to the example in that the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan. However, this change is more conservative than the example, in that the probability and consequences of previously-analyzed accidents are not increased and safety margins are not reduced.

A.(Continued)

3. Justification:

The proposed change is to clarify the surveillance requirement in accordance with the test method outlined in ANSI N45.4-1972 Appendix C.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples, (vi) of actions not likely to involve a significant hazards consideration is a change which may result in some increase to the probability or consequences of a previously-analyzed accident but that is clearly within all acceptance criteria in the Standard Review Plan.

The proposed change is similar to the example in that the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. However, this change is more conservative than the example, in that the probability and consequences of previously-analyzed accidents are not increased and safety margins are not reduced with the the rewording of Surveillance Requirements 4.6.1.2c.1 and 4.6.1.2c.3.

ATTACHMENT 9

CONTAINMENT LEAKAGE SPECIFICATION

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to change Surveillance Requirement 4.6.1.2c Items 1 and 3 to clarify that the superimposed leak, L_o, is used in the calculation and include new wording to comply with the Standard Technical Specifications and ANSI N45.4-1972.

1. Present condition of license:

As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Surveillance Requirements 4.6.1.2c.1 (page 6-3) that states: "Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a or 0.25 L_t.

As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Surveillance Requirement 4.6.1.2c.3 (page 6-3) that states: "Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at either greater than or equal to P_a, 47 psig, or greater than or equal to P_t, 25.0 psig, or at least four times the measured leakage in any one hour period."

2. Proposed condition of license:

Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to revise Surveillance Requirement 4.6.1.2c.1 (page 6-3) to state: "Confirms the accuracy of the Type A test by verifying that the supplemental test results, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a or 0.25 L_t, as applicable.

Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to revise Surveillance Requirement 4.6.1.2c.3 (page 6-3) to state: "Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a or 0.75 L_t and 1.25 L_t, as applicable."

Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) in LCO 3.6.1.2a.1 (page 6-2) and in Surveillance Requirements 4.6.1.2a, b, and c (page 6-3) insert an "as applicable" to provide flexibility for use of maximum allowable leakage rates L_a or L_t.

ATTACHMENT 10

VENTILATION SYSTEMS

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to update references to ANSI Standard N510-1975 in various surveillance requirements to the more recent version, N510-1980.

1. Present condition of license:

   As described in the applicable pages of the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102). This change would affect Sections 4.7.5, "Control Room Ventilation System" (pages 7-14 and 7-15), 4.7.6, "Auxiliary Building Safeguards Air Filtration System" (page 7-17), 4.9.12, "Fuel Handling Building Ventilation System" (pages 9-14 and 9-15), and their related Bases (pages B7-4 and B9-3), that presently have references to the 1975 version of ANSI N510.

2. Proposed condition of license:

   Change as described in the marked-up copy of the Technical Specifications (Attachment 22) to revise Surveillance Requirements 4.7.5.1c.3 (page 7-14), 4.7.5.1e.4 (page 7-15), 4.7.5.1f (page 7-15), 4.7.5.1g (page 7-15), 4.7.6.1b.4 (page 7-17), 4.7.6.1d.3 (page 7-17), 4.7.6.1e (page 7-17), 4.7.6.1f (page 7-17), 4.9.12b.4 (page 9-14), 4.9.12e (page 9-14), and 4.9.12f (page 9-15), and Bases 3/4.7.5 (page B7-3), 3/4.7.6 (page B7-4) and 3/4.9.12 (page B9-3), to reference ANSI Standard N510-1980 rather than N510-1975.

3. Justification:

   The changes described are to update ventilation system Technical Specification surveillance requirements to reference a more recent version of ANSI Standard N510.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

B.(Continued)

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (i) of actions not likely to involve a significant hazards consideration is a purely administrative change to the Technical Specifications. These proposed changes are administrative in nature only. The 1980 version of ANSI Standard N510 reflects a clarification of test requirements without changing the acceptance criteria, thus the change is administrative in nature and does not involve a significant hazards consideration.

ATTACHMENT 11

FIRE SUPPRESSION WATER VALVES

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to address the surveillance frequency for verification of several fire suppression water valves inside containment. The amendment would add a footnote to Surveillance Requirement 4.7.9.1c that allows locked or sealed fire suppression water valves inside containment to be verified in the correct position during each Cold Shutdown rather than "at least once per 31 days."

1. Present condition of license:

As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Surveillance Requirement 4.7.9.1c (page 7-27), which states that the Fire Suppression Water System shall be demonstrated operable "at least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position."

2. Proposed condition of license:

Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to add a footnote to Surveillance Requirement 4.7.9.1c (page 7-27) to qualify the word "valves" by stating "*Except valves which are located inside the containment and are locked, sealed, or otherwise secured in position. These valves shall be verified in the correct position during each Cold Shutdown except such verification need not be performed more often than once per 92 days."

3. Justification:

The addition of this footnote is similar to the footnote on Surveillance Requirement 4.6.1.1a (page 6-1) for Containment Integrity that allows less frequent verification of those valves "which are located inside the containment and are locked, sealed, or otherwise secured in the closed position." The reason for this qualification is due to ALARA considerations. Due to the limited access into the containment and the locking or sealing of these valves (for documentation and insurance) in the correct position, there is sufficient confidence that the increase in the surveillance interval will not compromise the systems operability. The overall benefit of this change will be the reduction of exposure to personnel who would have had to make a containment entry at power to perform the verification for Surveillance Requirement 4.7.9.1c.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (vi) of actions not likely to involve a significant hazards consideration is a change which may result in some increase to the probability or consequences of a previously-analyzed accident but that is clearly within all acceptable criteria in the Standard Review Plan.

The proposed change fits this example in that a relaxation in the surveillance interval in itself may result in a slight increase in the probability of unavailability of Fire Suppression Water to components in the containment. However, compensatory measures shall be taken to lock or seal these valves in the correct position to ensure availability. With this offsetting action the net benefit for this change is the reduction in radiation exposure to operating personnel performing the surveillance test (ALARA consideration). The change is minor and does not affect the acceptance criteria for the Fire Suppression Water System as described in the Standard Review Plan.

ATTACHMENT 12

SPRAY AND/OR SPRINKLER SYSTEMS

A. Description of amendment request:

This amendment revises the Technical Specification of the Operating License to add additional areas to Specification 3/4.7.9.2, "Spray and/or Sprinkler Systems" (page 7-28).

1. Present condition of license:

   As described in the Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), the Limiting Condition for Operation of Specification 3.7.9.2, "Spray and/or Sprinkler Systems" (page 7-28), lists those areas and equipment for which spray and/or sprinkler systems shall be operable.

2. Proposed condition of license:

   Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to add the following additional areas to the Limiting Condition for Operation of Specification 3.7.9.2 "Spray and/or Sprinkler Systems" (page 7-28): "f. Centrifugal Charging Pumps Area" and "h. Containment Penetration Area". Also Specification 3.7.9.2e shall be revised to say "e. Auxiliary Feed Pumps Area."

3. Justification:

   The changes described are to update the list of areas required to have spray and/or sprinkler systems operable to provide equipment protection.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (ii) not likely to involve a significant hazards consideration is a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications; for example, a more stringent surveillance requirement.

The proposed changes are similar to this example in that an additional restriction is imposed by requiring additional areas to be protected by spray and/or sprinkler systems.

ATTACHMENT 13

FIRE BARRIER PENETRATION

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to change the applicability of the Fire Barrier Penetration Technical Specification in accordance with staff guidance on 10 CFR 50 Appendix R.

1. Present condition of license:

   As described in the Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), the applicability for Specification 3.7.10, "Fire Barrier Penetrations" (page 7-36), is "at all times."

2. Proposed condition of license:

   Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to make the applicability of Technical Specification 3.7.10, "Fire Barrier Penetrations" (page 7-36), "Whenever the equipment protected by the fire barrier penetration is required to be OPERABLE."

3. Justification:

   The change is to update the fire barrier penetration specification to agree with the current NRC staff position presented in the April 5, 1984 Regional Appendix R Workshop.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the possibility or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (vi) not likely to involve a significant hazards consideration is a change which may either result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan.

B.(Continued)

The proposed change is similar to the example in that the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. However, this change is more conservative than the example, in that the probability and consequences of previously-analyzed accidents are not increased and safety margins are not reduced since the inoperability of the fire barrier penetrations will not affect the operability of the equipment protected, if the equipment is not required to be operable.

ATTACHMENT 14

DELETED

ATTACHMENT 15

DIESEL FUEL OIL SPECIFICATION

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to incorporate new Surveillance Requirements for emergency diesel fuel oil. The proposed change is designed to provide improvement in the level of confidence for the quality of the diesel fuel oil and is consistent with Unit 2 Technical Specifications.

1. Present condition of license:

   As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Specification 4.8.1.1.3 (page 8-6 and 8-7), outlines the testing requirements for the emergency diesel fuel oil.

2. Proposed condition of license:

   Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to incorporate new surveillance requirements for emergency diesel fuel oil, Specification 4.8.1.1.3 (page 8-6 and 8-7).

3. Justification:

   The changes described revise Specification 4.8.1.1.3 (page 8-6 and 8-7) to be consistent with draft Revision 5 of the Westinghouse Technical Specifications and the most recent NRC staff position.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (ii) relates to changes that constitute an additional limitation, restriction, or control not presently included in the Technical Specifications; for example, a more stringent surveillance requirement.

B.(Continued)

The proposed change provides new surveillance requirements for the diesel fuel oil that are consistent with the NRC Staff present position related to testing of diesel fuel oil. The proposed surveillance requirements are designed to provide improvement in the level of confidence in the quality of the diesel fuel oil and to assure that the diesel fuel oil is maintained within the acceptance criteria included in Specification 4.8.1.1.3.

ATTACHMENT 16

FREQUENCY OF BATTERY TEST

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to change the frequency for performing the battery capacity performance discharge test for Specification 3/4.8.3.1.

1. Present condition of license:

   As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Surveillance Requirement 4.8.3.1f (page 8-15) requires the battery capacity performance discharge test to be performed annually.

2. Proposed condition of license:

   Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to revise Surveillance Requirement 4.8.3.1f (page 8-15) to say "At least once per 18 months during shutdown by performing a performance discharge test ..."

3. Justification:

   The proposed change allows for a more realistic time interval to perform this test since the batteries are required to be operable during Modes 1, 2, 3 and 4 and the batteries must be removed from service for the performance of this test. This change is consistent with draft Revision 5 of the Westinghouse Standard Technical Specifications.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (vi) of actions not likely to involve a significant hazards consideration is a change which may result in some increase to the probability or consequences of a previously-analyzed accident but that is clearly within all acceptable criteria in the Standard Review Plan.

B.(Continued)

The proposed change is similar to this example in that the surveillance interval to perform the battery capacity performance discharge test has been increased slightly to allow the test to be performed when the batteries are not required to be operable. However, the change in time is minor and does not affect the acceptance criteria for the batteries as described in the Standard Review Plan and is consistent with draft Revision 5 of the Westinghouse Standard Technical Specifications.

ATTACHMENT 17

ADD 3.0.3/3.0.4 TO REFUELING SPECIFICATIONS

A. Description of amendment request:

This amendment revises the Technical Specifications to conform with the current NRC Staff position on various Refueling Operation specifications. These changes make Specification 3.0.3 and/or 3.0.4 not applicable in several situations not related to reactor operation and are consistent with draft Revision 5 to the Westinghouse Standard Technical Specifications. PGandE was advised of this staff position in meetings with the NRC.

1. Present condition of license:

As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), the Action Statements of Specification 3.9.7, "Crane Travel-Fuel Handling Building" (page 9-7), 3.9.9, "Containment Ventilation Isolation System" (page 9-10), 3.9.11, "Water Level-Spent Fuel Pool" (page 9-12), and 3.9.12, "Fuel Handling Building Ventilation System" (page 9-13).

2. Proposed condition of license:

Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to state that Specifications 3.0.3 and 3.0.4 are not applicable to Specifications 3.9.7, "Crane Travel-Fuel Handling Building" (page 9-7), 3.9.9, "Containment Ventilation Isolation System" (page 9-10), and 3.9.11, "Water Level-Spent Fuel Pool" (page 9-12); and Specification 3.0.3 is not applicable to Specification 3.9.12, "Fuel Handling Building Ventilation System" (page 9-13).

3. Justification:

The changes described are to revise the Refueling Operation Specifications to conform to the current staff positions presented in meetings with the licensee and are consistent with draft Revision 5 to the Westinghouse Standard Technical Specifications.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (vi) not likely to involve a significant hazards consideration is changes

B.(Continued)

which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan.

The proposed change is similar to the example in that the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan and is consistent with draft Revision 5 of the Westinghouse Standard Technical Specifications. However, this change is more conservative than the example, in that the probability and consequences of previously-analyzed accidents are not increased and safety margins are not reduced. The revised specifications are concerned with the spent fuel pool and not with reactor operation. Therefore, Specifications 3.0.3 and 3.0.4 that deal with reactor operations are not applicable to these specifications.

ATTACHMENT 18

ORGANIZATIONAL CHANGES

A. This amendment revises the Technical Specifications of the Operating License to incorporate various organizational changes to Section 6.0, "Administrative Controls",

1. Present condition of license:

   As described in the current Diablo Canyon Unit 1 Technical Specification (NUREG-1102) Section 6.0, "Administrative Controls", Subsection 6.1, "Responsibility" (page 6-1), Subsection 6.2, "Organization" (pages 6-1 through 6-6), Subsection 6.3, "Unit Staff Qualifications" (page 6-6), Subsection 6.4, "Training" (page 6-6), Subsections 6.5.1.6 and 6.5.1.7 "Plant Staff Review Committee (PSRC) Responsibilities" (pages 6-7 and 6-8), Subsection 6.5.1.7, "PSRC Authority" (page 6-8), Subsection 6.5.1.8, "PSRC Records" (page 6-8), Subsection 6.5.2, "General Office Nuclear Plant Review and Audit Committee (GONPRAC)" (pages 6-9 through 6-12), Subsection 6.6, "Reportable Occurrences Action" (page 6-12), Subsection 6.7, "Safety Limit Violation" (page 6-12), Subsection 6.9.1.4, "Annual Reports" (page 6-15), Subsection 6.9.1.6, "Annual Radiological Environmental Operating Report" (page 6-16) and Subsection 6.9.1.8, "Semi-annual Radioactive Effluent Release Report" (page 6-17).

2. Proposed condition of license:

   Changes as described in Subsections 6.1 through 6.5 and 6.9 of the marked-up copy of the Technical Specifications (Attachment 22) to reflect a plant organization as opposed to a unit organization.

   Changes as described in the marked-up copy of the Technical Specifications to the titles of various members of the organization.

   On Figure 6.2-1, "Offsite Organization" (page 6-2), the title "Technical Assistant to Vice President Nuclear Power Generation" is revised to "Manager, Nuclear Operations Support"; the title "Quality Assurance Engineer" is revised to "Director Quality Support"; the title "Manager, Nuclear Plant Operations" has been deleted and "On Site Organization" has been revised to "Plant Organization".

   On Figure 6.2-2, "Unit Organization" (page 6-3) the title "Quality Assurance Engineer has been revised to "Director Quality Support"; the title "Tech. Asst. to VP Nuclear Power Generation" has been revised to "Manager, Nuclear Operations Support" and the figure title revised from "Unit Organization" to "Plant Organization."

   In Subsection 6.5.1.6, 6.5.1.7 and 6.5.1.8, "Plant Staff Review Committee (PSRC) Responsibilities" (page 6-8), the title "Manager of Nuclear Plant Operations" has been revised to "Vice President, Nuclear Power Generation." Reference to "Chairman of GONPRAC" has been deleted.

A.(Continued)

In subsection 6.2.3, "Onsite Safety Review Group (OSRG)" (page 6-5), the title "Technical Assistant to the Vice President, Nuclear Power Generation" has been revised to "Manager, Nuclear Operations Support".

In subsection 6.5.2, "General Office Nuclear Plant Review and Audit Committee (GONPRAC)" (pages 6-9 through 6-11), the GONPRAC Authority paragraph located on page 6-12 has been moved to Section 6.5.2.1 on page 6-9. Under "Composition 6.5.2.2" the title "Manager, Nuclear Plant Operations" has been revised to "Manager, Nuclear Operations Support," and one new member has been added with the title "Assistant to the Vice President, Nuclear Power Generation" as Vice Chairman. The footnote at the bottom of the page describing qualifications for two revised titles has been deleted.

Under "Review" Subsection 6.5.2.7, item "i" regarding "Reports and Meeting Minutes of the Plant Staff Review Committee" the words "and the Onsite Safety Review Group" have been added.

In subsection 6.6 and 6.7, "Reportable Event Action" and "Safety Limit Violation" (page 6-12), the title "Manager of Nuclear Plant Operations" has been revised to "Vice President, Nuclear Power Generation" and "Executive Vice President, Facilities and Electric Resources Development," respectively

3. Justification:

These changes are necessary to reflect a plant organization and PGandE's reorganization of Nuclear Power Generation. The changes provide for direct reporting of the Plant Manager to the Vice President, Nuclear Power Generation and also result in more operational experience for the GONPRAC organization.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (i) of actions not likely to involve a significant hazards consideration is a purely administrative change to the Technical Specifications. The proposed changes fit this example in that all of the changes are administrative in nature and do not change the technical content of the specifications.

ATTACHMENT 19

RHR SUCTION ISOLATION VALVES

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to delete the requirement to lock open the Residual Heat Removal (RHR) Suction Isolation Valves, 8701 and 8702, when the positive displacement charging pump is in operation and the unit is in "Mode 4" with the temperature of any RCS cold leg less than or equal to 323°F, or when in Mode 5 and Mode 6 with the reactor vessel head on.

1. Present condition of license:

As described in the current Diablo Canyon Unit 1 Technical Specification (NUREG-1102), Specification 3/4.4.9.3, "Overpressure Protection Systems", Limiting Condition for Operation 3.4.9.3a (page 4-34) that states, "RHR system isolation valves 8701 and 8702 open with power removed from the valve operators when the positive displacement charging pump is in operation." Action a (page 4-34) that states, "with the positive displacement charging pump in operation with the RHR isolation valves closed, within one hour either open the RHR isolation valves or secure the positive displacement charging pump," Surveillance Requirement 4.4.9.3.1.d (page 4-35) that states, "Testing pursuant to Specification 4.0.5", and Surveillance Requirement 4.4.9.3.1.e (page 4-35) that states, "Verifying the RHR isolation valves 8701 and 8702 are opened with power removed from the valve operators when the positive displacement charging pump is in operation at least once per 72 hours."

2. Proposed condition of license:

Changes as described in the marked-up copy of the Technical Specifications (Attachment 22) to delete Limiting Condition for Operation 3.4.9.3a (page 4-34) and create a new Limiting Condition for Operation 3.4.9.3a by combining the existing Limiting Condition for Operation 3.4.9.3b and 3.4.9.3c, delete Action a, delete Surveillance Requirement 4.4.9.3.1.d and delete Surveillance Requirement 4.4.9.3.1.e.

3. Justification:

The change described is to conform with NRC requirements as described in Diablo Canyon SSER-21.

B. Proposed basis for no significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

B.(Continued)

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of the examples (iv) of action not likely to involve a significant hazards consideration is a relief granted upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation was not yet demonstrated when the operating restriction and the criteria to be applied to a request for relief have been established in a prior review and it is justified in a satisfactory way that the criteria have been met. Diablo Canyon SSER-21 authorizes removal of this Action Statement following installation of an RHR low flow alarm. The RHR flow flow alarm has been installed and is functional for both Unit 1 and Unit 2.

ATTACHMENT 20

UNIT 2 TECHNICAL SPECIFICATIONS APPLICABLE TO UNIT 1

A. Description of amendment request:

This amendment revises the Technical Specifications of the Operating License to include those changes necessary to create a combined Unit 1 and 2 Technical Specification. These changes include: 1) those Technical Specifications required for the creation of a common document, 2) revisions to the Unit 1 Technical Specifications to reflect changes made to the Unit 2 Technical Specifications during the NRC Staff review process, and 3) those outstanding License Amendment Requests on Unit 1 that were included in the issued Unit 2 Technical Specifications.

1. Present condition of license:

As described in the current Diablo Canyon Unit 1 Technical Specifications (NUREG-1102), Sections 3.0 and 4.0, "Limiting Conditions for Operation and Surveillance Requirements" (pages 0-1 and 0-3), Specification 3/4.5.1, "Accumulators" (pages 5-1 and 5-2), Specification 3/4.6.1.4, "Containment Systems - Internal Pressure" (page 6-7), Specification 3/4.8, "Electrical Power Systems" (pages 8-1 through 8-21), Table 3.3-5, "Engineered Safety Features Response Times" (page 3-31), Specification 3/4.3.3.6, "Accident Monitoring Instrumentation" (pages 3-51, 3-52 and 3-53), and Bases 6.1.4, "Internal Pressure" (page B 3/4 6-1).

2. Proposed condition of license:

Changes as described in the marked-up copy of the Technical Specifications, (Attachment 22) to add new Specifications 3.0.5 and 4.0.6 (pages 0-1 and 0-3), to revise Limiting Condition for Operation 3.5.1a and Surveillance Requirement 4.5.1.1d (pages 5-1 and 5-2) regarding accumulator isolation valve testing, to revise the Limiting Condition for Operation 3.6.1.4 to reflect a new minimum containment internal pressure (page 6-7), to add a new Specification 3/4.8.4.2 "Containment Penetration Conductor Overcurrent Protective Devices (pages 8-22 through 8-26)," and to incorporate three outstanding License Amendment Requests by adding the Feedwater Bypass valves and closure times to "Notation (2)" of Table 3.3-5 "Engineered Safety Features Response Times" (page 3-31), adding the Reactor Vessel Level Indication System to Specification 3/4.3.3.6 "Accident Monitoring Instrumentation" (pages 3-51, 3-52 and 3-53), and revising Bases 6.1.4 "Internal Pressure" (page B 3/4 6-1) to reflect changes made to containment spray timing.

3. Justification:

The changes described are to: 1) Add new Specifications 3.0.5 and 4.0.6 that are required to form a combined Unit 1 and 2 Technical Specification document. These Specifications provide guidance for the use of the combined Technical Specifications and applicability of the Limiting Conditions For Operation and Surveillance Requirements. 2) Revise several Specifications to be in conformance with the NRC Staff positions incorporated in the recently issued Unit 2 Technical Specifications. These positions included a change in the value for the Containment Negative Pressure Limit, the addition of a new Specification for Containment Penetration Conductor Overcurrent Protective Devices and changes to the Accumulator Specification. 3) Add several outstanding License Amendment Requests that were included in the Unit 2 Technical Specifications. These License Amendment Requests are LAR 84-10 that added Feedwater Bypass valves to the Technical Specifications, LAR 84-11 that added the Reactor Vessel Level Indication System to the Technical Specifications and to correct an oversight in the issuance of the change requested by LAR 83-06 that was inadvertently left out of the License Amendment.

B. Proposed basis for not significant hazards consideration determination:

The standards used to arrive at a proposed determination that a request for amendment involves no significant hazards consideration are included in 10 CFR 50.92. The regulations state that the operation of the facilities in accordance with the proposed amendment would not: 1) involve a significant increase in the probability or consequences of an accident previously evaluated, or 2) create the possibility of a new or different kind of accident from any accident previously evaluated, or 3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on April 6, 1983. One of these examples (vi) of actions not likely to involve a significant hazards consideration is a change which may result in some increase to the probability or consequences of a previously-analyzed accident but that is clearly within all acceptable criteria in the Standard Review Plan.

The proposed changes are similar to this example in that these changes are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. The NRC Staff has reviewed, approved, and incorporated these changes, expect Specifications 3.0.5 and 4.0.6 into the Unit 2 Technical Specifications. Specifications 3.0.5 and 4.0.6, which provide guidance for the use of combined Technical Specifications, have been approved by the NRC Staff for combined Technical Specifications.

ATTACHMENT 21

UNIT 2 DIFFERENCES

The common Technical Specifications for Diablo Canyon Unit 1 and 2 are based upon the current Unit 1 Technical Specifications (NUREG-1102), however, a few minor revisions were required to make the Unit 1 Technical Specifications applicable to both units. The attached marked-up copy of the Technical Specifications (Attachment 22) includes the Unit 2 differences described below.

The final product is a common Diablo Canyon (both Units 1 and 2) Technical Specification, with the bottom of all pages corrected to reflect this.

1. Definition 1.28, "Rated Thermal Power" (page 1-5), was revised to include the Unit 2 power level of 3411 MWt.

2. The footnote on Item 12 of Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints" (page 2-4), was revised to include the Unit 2 loop design flow. The value for Unit 2 is 88,500 gpm which is slightly higher than Unit 1 due to differences in Unit 2 reactor internal characteristics which result in decreased head losses.

3. The value for reference T_{AVG} in Notes 1 and 3 of Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints" (pages 2-7 and 2-9), was revised to include the Unit 2 value of 577.6°F. The difference is due to the higher rated thermal power and loop design flow of Unit 2.

4. The value for the steam/feedwater flow mismatch described in the Limiting Safety System Settings Bases was revised to include the Unit 2 setpoint of 1.49×10^6 lbs/hour (page B2-7). The difference is due to the higher rated thermal power of Unit 2.

5. Figure 3.1-1b, "Rod Bank Insertion Limit Versus Thermal Power" (page 1-24), was added to include the rod insertion limits for Unit 2. The higher rod insertion limits are due to the higher rated thermal power of Unit 2 which requires more available negative reactivity to meet the shutdown margin requirements.

6. Figure 3.2-3b, "RCS Total Flowrate Versus R" (page 2-13), was added to include the different limits for Unit 2. The difference is due to the higher loop design flow of Unit 2.

7. The value for Reactor Coolant System T_{AVG} in Table 3.2-1, "DNB Parameters" (page 2-18), was revised to include the Unit 2 value of 582°F. The difference is due to the higher rated thermal power and loop design flow of Unit 2.

8. Item 1.b, "Control Room Ventilation Mode Change", of Table 3.3-6, "Radiation Monitoring Instrumentation" (page 3-38), was revised and a footnote was added to clarify the minimum channels operable requirement of the monitors on the common Control Room Ventilation System.

9. A footnote was added to Specification 3.3.3, "Seismic Instrumentation" (page 3-42), to reflect that the Seismic Monitoring Instrumentation is common to both units.

10. A footnote was added to Specification 3.3.4, "Meteorological Instrumentation" (page 3-45), to reflect that the Meteorological Monitoring Instrumentation channels are common to both units.

11. A footnote was added to Specification 3.3.7, "Chlorine Detection Systems" (page 3-54), to reflect that the Chlorine Detection Systems are common to both units.

12. Some instrument locations on Table 3.3-11, "Fire Detection Instruments - Panel B" (page 3-57), were revised to include Unit 2 locations. Also, a footnote was added to this table (page 3-58) to reflect that fire pumps and Diesel Generator No. 3 are common to both units.

13. A footnote was added to Table 3.3-12, "Radioactive Liquid Effluent Monitoring Instrumentation" (page 3-60), to reflect that instruments 1.a, "Liquid Radwaste Effluent Line Radioactivity Monitor (RM-18)", 2.a, "Liquid Radwaste Effluent Line Flow Rate Device (FR-18)", 2.c, "Oily Water Separator Effluent Line Flow Rate Device (FR-251)" and 3.a, "Oily Water Separator Effluent Line Radiation Monitor (RM-3)" are common to both units.

14. Table 4.4-5, "Reactor Vessel Material Surveillance Program - Withdrawal Schedule" (page 4-30), was revised to include the Unit 2 values.

15. The Containment Firewater Isolation Check Valve number for Unit 2 was added to Table 3.6-1, "Containment Isolation Valves" (page 6-24), since the valve number is different from Unit 1.

16. A footnote was added to Specification 7.5, "Control Room Ventilation System" (page 7-13), to reflect that the Control Room Ventilation System is common to both units.

17. A note was added to Item c., "Diesel Generator No. 3 Flooding System", of Table 3.7-3, "CO_2 System" (page 7-31), that states that this system is common to both units.

18. Item e, "CO_2 Hose Reel Subsystem Stations", of Table 3.7-3, "CO_2 System" (page 7-31), was revised to include the Unit 2 hose reel identification numbers.

19. Table 3.7-4, "Fire Hose Stations" (pages 7-34 and 7-35), was revised to include the locations and hose station identifications for Unit 2.

20. A note was added to Item 15, "Diesel Generator No. 3 Room", of Table 3.7-5, "Area Temperature Monitoring" (page 7-38), that states that this area is common to both units.

21. A footnote was added to Specification 7.12, "Ultimate Heat Sink" (page 7-39), to reflect that the Ultimate Heat Sink is common to both units.

22. A footnote was added to Specification 7.13, "Flood Protection" (page 7-40), to reflect that the Breakwaters (east and west) are common to both units.

23. A footnote was added to Specification 9.8.2, "Refueling Operations - Low Water Level" (page 9-9), for Residual Heat Removal (RHR) Pump operability during refueling operations when the water level above the top of the reactor vessel flange is less than 23 feet. The footnote allows for core alterations in the vicinity of the reactor vessel hot legs with the RHR pumps secured for up to 1 hour per 8 hour period. This relaxation is only applicable prior to initial criticality and therefore is only applicable to Unit 2.

24. Specification 9.13 "Spent Fuel Shipping Cask Movement" (page 9-16), was revised to add a Unit 2 reference to the Limiting Condition for Operation. The reference is in regard to Spent Fuel shipping cask movement near the spent fuel pool and states "... or south of column line 23.1 for Unit 2".

25. A footnote was added to Specification 11.1.3, "Liquid Radwaste Treatment System" (page 11-6), to reflect that the Liquid Radwaste Treatment System is common on both units.

26. Table B 3/4.4-1a (page B4-8) was revised by adding Unit 1 to the title. Tables B 3/4.4-1b, B 3/4.4-1C and B 3/4.4-1d (pages B4-9 and B4-10) were deleted and substituted by a new table B 3/4.4-1b, "Reactor Vessel Toughness Data-Unit 2" (page B4-9), which includes the specific data for Unit 2.

27. The Reactor Coolant System Volume described in Specification 5.4.2, "Volume" (page 5-5), was revised to reflect the Unit 2 value.

28. Subsection 6.5.2, "General Office Nuclear Plant Review and Audit Committee (GONPRAC)" (pages 6-10 through 6-12) of Section 6.0 "Administrative Controls" was revised to reflect two unit operation by changing the word "unit" to "plant" in several instances.

ATTACHMENT 22

MARKED UP TECHNICAL SPECIFICATIONS

This attachment provides the entire Unit 1 Technical Specification (Appendix A to Operating License DPR-80) marked up consistent with the Unit 2 Technical Specifications.

NUREG-1102

DIABLO CANYON NUCLEAR POWER STATION

UNIT 1

TECHNICAL SPECIFICATIONS

APPENDIX "A"

TO

LICENSE NO. DPR-80

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NOT MARKED UP

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ADMINISTRATIVE CONTROLS

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

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

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or trip setpoints such that the setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

SAME ON ALL PAGES

CHANNEL FUNCTIONAL TEST

1.7 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions, or

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is ~~OPERABLE pursuant to~~ in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.9 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.10 CORE ALTERATIONS shall be the movement or manipulation of any component within the reactor ~~pressure~~ vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position.

DEFINITIONS

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/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," or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average ~~sum~~ (weighted in proportion to the concentration of each radionuclide in the SAMPLE ~~reactor coolant at the time of sampling~~) of the sum of the average beta and gamma energies per disintegration (~~in MeV~~ MeV/d) ~~for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant~~ for the radionuclides in the sample.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURE

1.14 The ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURE (ERMP) shall be contained in PG&E's Department of Engineering Research Environmental Radiological Monitoring Manual. It shall contain a description of sample locations, types of sample locations, methods and frequency of analysis, and reporting requirements.

FREQUENCY NOTATION

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table ~~1.2~~ 1.1.

GASEOUS RADWASTE SYSTEM

1.16 A GASEOUS RADWASTE SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting ~~primary~~ Reactor Coolant System offgases from the ~~primary~~ System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.17 IDENTIFIED LEAKAGE shall be:

a. Leakage, except CONTROLLED LEAKAGE, into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of ~~L~~eakage ~~d~~etection ~~S~~ystems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the ~~S~~econdary Coolant ~~S~~ystem.

MASTER RELAY TEST

1.18 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

1.19 Members of the Public

OFFSITE DOSE CALCULATION PROCEDURE

1.~~19~~ 20 The OFFSITE DOSE CALCULATION PROCEDURE (ODCP) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring ~~A~~larm/~~T~~rip ~~S~~etpoints.

OPERABLE - OPERABILITY

1.~~20~~ 21 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electric power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

Insert

1.19 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 licensee, 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.

PROCESS CONTROL PROGRAM

1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and Federal and State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste. Insert

OPERATIONAL MODE - MODE

1.~~21~~ 22 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.~~1~~ 2.

PHYSICS TESTS

1.~~22~~ 23 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission. (3)

PRESSURE BOUNDARY LEAKAGE

1.~~23~~ 24 PRESSURE BOUNDARY LEAKAGE shall be leakage, except steam generator tube leakage, through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM

(1)

1.~~24~~ 25 ~~The PROCESS CONTROL PROGRAM (PCP) shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.~~

PURGE - PURGING

1.~~25~~ 26 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.~~26~~ 27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.~~27~~ 28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3338 Mwt for Unit 1 and 3411 Mwt for Unit 2.

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from
29 when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE ~~OCCURRENCE~~ EVENT

1.29 A REPORTABLE ~~OCCURRENCE~~ EVENT shall be any of those conditions specified in
30 ~~Specifications 6.9.1.12 and 6.9.1.13~~. SECTION 50.73 of 10 CFR PART 50

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which
31 the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.32 SITE BOUNDARY shall be that line as shown in Figure 5.1-3.

SLAVE RELAY TEST

1.31 A SLAVE RELAY TEST shall be the energization of each slave relay and
33 verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.32 SOLIDIFICATION shall be the conversion of ~~radioactive~~ WET wastes ~~from liquid~~
34 ~~systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing)~~. into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.33 A SOURCE CHECK shall be the qualitative assessment of channel response
35 when the channel sensor is exposed to a ~~radioactive~~ source, or increased radioactivity.

STAGGERED TEST BASIS

1.34 A STAGGERED TEST BASIS shall consist of:
36

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

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

DEFINITIONS

THERMAL POWER

1.~~25~~ 37 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.~~26~~ 38 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.~~27~~ 39 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

40

VENTILATION EXHAUST TREATMENT SYSTEM

1.~~28~~ 41 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. (Such a system is not considered to have any effect on noble gas effluents). Engineered Safety Features ~~(ESF)~~ Atmospheric Cleanup Systems are not normally considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.~~29~~ 42 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

UNRESTRICTED AREA

Insert

1.40 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 350°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 350°F$
3. HOT STANDBY	< 0.99	0	$\geq 350°F$
4. HOT SHUTDOWN	< 0.99	0	$350°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6. REFUELING**	≤ 0.95	0	$\leq 140°F$

* Excluding decay heat.

** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Completed prior to each release
N.A.	Not applicable.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of specification 6.7.

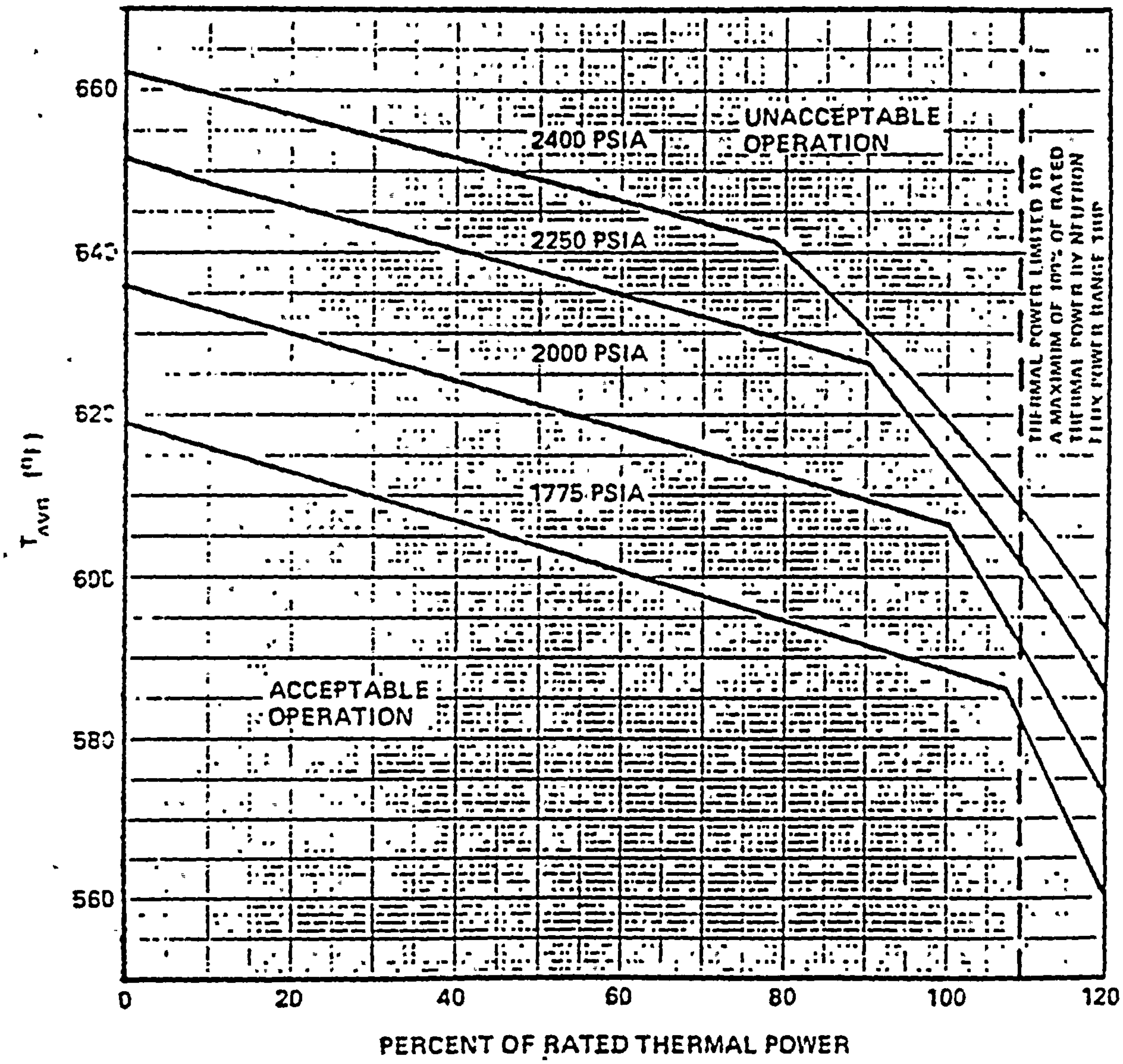

FIGURE 2.1-1

REACTOR CORE SAFETY LIMITS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. Low Setpoint	≤ 25% of RATED THERMAL POWER	≤ 26% of RATED THERMAL POWER
b. High Setpoint	≤ 109% of RATED THERMAL POWER	≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3×10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 2
8. Overpower ΔT	See Note 3	See Note 4
9. Pressurizer Pressure--Low	≥ 1950 psig	≥ 1940 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Reactor Coolant Flow--Low	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

*Design flow is 87,700 gpm per loop for Unit 1 and 88,500 gpm per loop for Unit 2.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 15% of narrow range instrument span-each steam generator	≥ 14% of narrow range instrument span-each steam generator
14. Steam Generator Water Level-Low Coincident with	≥ 25% of narrow range instrument span--each steam generator	≥ 24% of narrow range instrument span--each steam generator
Steam/Feedwater Flow Mismatch	< 40% of full steam flow at RATED THERMAL POWER	≤ 42.5% of full steam flow at RATED THERMAL POWER
15. Undervoltage-Reactor Coolant Pumps	≥ 8050 volts-each bus	≥ 7935 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 54.0 Hz - each bus	≥ 53.9 Hz - each bus
17. Turbine Trip		
a. Low Autostop Oil Pressure	≥ 50 psig	≥ 45 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
18. Safety Injection Input ~~from~~ from ESF	N.A.	N.A.
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.
20. Reactor Trip Breakers	N.A.	N.A.
21. Automatic Trip and Interlock Logic	N.A.	N.A.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
22. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7		
a. P-10 Input	10% of RATED THERMAL POWER	> 9%, < 11% of RATED THERMAL POWER
b. P-13 Input	< 10% RTP Turbine Impulse Pressure Equivalent	< 11% RTP Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	< 35% of RATED THERMAL POWER	< 36% of RATED THERMAL POWER
d. ~~Low Setpoint~~ Power Range Neutron Flux, P-10	10% of RATED THERMAL POWER	> 9%, < 11% of RATED THERMAL POWER
e. Turbine Impulse Chamber Pressure, P-13	< 10% RTP Turbine Impulse Pressure Equivalent	< 11% RTP Turbine Impulse Pressure Equivalent
23. Seismic Trip	$\leq$ 0.35 g	$\leq$ 0.40 g

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

NOTE 1: ~~OVERTEMPERATURE~~ $\Delta T \leq \Delta T_0 \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)(T - T') + K_3(P - P') - f_1(\Delta I)\right]$

Where: ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

T' = $\leq$ 576.6°F for Unit 1 and $\leq$ 577.6°F for Unit 2 Reference T_{avg} at RATED THERMAL POWER;

P = Pressurizer pressure, psig;

P' = 2235 psig (indicated RCS nominal operating pressure);

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 30 secs, τ_2 = 4 secs;

S = Laplace transform operator, sec^{-1};

K_1 = 1.174;

K_2 = 0.01358/°F;

K_3 = 0.000685/psig;

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS (Continued)

NOTE 1 (Continued)

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 32 % ~~percent~~ and + 10 % ~~percent~~, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -32 % ~~percent~~, the ΔT Trip Setpoint shall be automatically reduced by 2.11 % ~~percent~~ of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds + 10 % ~~percent~~, the ΔT Trip Setpoint shall be automatically reduced by 1.45 % ~~percent~~ of its value at RATED THERMAL POWER.

Note 2: The channel's maximum Trip Setpoint ~~point~~ shall not exceed its computed Trip Setpoint ~~point~~ by more than 4 percent.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS (Continued)

Note 3: Overpower $\Delta T \leq \Delta T_0 \left[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 \left(T - T''\right) - f_2(\Delta I)\right]$

Where: ΔT_0 = Indicated ΔT at rated power;

T = Average temperature, °F;

T'' = $\leq$ 576.6°F for Unit 1 and $\leq$ 577.6°F for Unit 2 Reference T_{avg} at RATED THERMAL POWER;

K_4 = 1.079;

K_5 = 0.0174/°F for increasing average temperature and 0 for decreasing average temperature;

K_6 = 0.00121/°F for $T > T''$; $K_6 = 0$ for $T < T''$;

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation;

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs;

S = Laplace transform operator, sec^{-1}; and

$f_2(\Delta I)$ = 0 for all ΔI.

Note 4: The channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than 3 percent.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

The BASES contained in the succeeding pages summarize the reasons ~~The summary statement contained in this section provides the bases~~ for the Specifications of Section 2.0, but in accordance with 10 CFR 50.36 are not ~~a~~ part of these Technical Specifications.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

R-Grid ③

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the ~~W-3~~ correlation. The ~~W-3~~ DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and 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. This value corresponds to a 95 ~~percent~~ probability at a 95 ~~percent~~ confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$ of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.2\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (~~delta~~ Δ I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ~~delta~~ Δ T trip will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor ~~pressure~~ vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping and fittings are designed to ANSI B 31.1, ~~1967 Edition~~ which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the Reactor ~~Protection~~ Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor ~~Protection~~ Trip System which monitors numerous system variables, therefore, providing protection Trip system functional diversity.

The Reactor ~~Protection~~ Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance for all trips including those trips assumed in the safety analyses.

Manual Reactor Trip

The Reactor ~~Protection~~ Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBR's will be greater than 1.30.

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10 ~~percent~~ % of RATED THERMAL POWER) and is automatically reinstated below the P-10 setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

~~The Power Range Negative Rate trip provides protecton to ensure that the minimum DNBR is maintained above 1.30 for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occuring by tripping the reactor for all single or multiple dropped rods.~~

Intermediate and Source Range, ~~Nuclear~~ Neutron Flux

The Intermediate and Source Range, ~~Nuclear~~ Neutron Flux trips provide ~~reactor~~ core protection during reactor STARTUP to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25 ~~percent~~ % of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor ~~Protection~~ Trip System.

Overpower ΔT

The Overpower ~~delta~~ Δ T trip provides assurance of fuel integrity, e.g., no fuel pellet cracking or melting, under all possible overpower conditions, limits the required range for Overtemperature ~~delta~~ Δ T protection, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied

Overpower ΔT (Continued)

with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change at temperature for dynamic compensation for piping delays from the core to the loop temperature detectors to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam ~~Break~~ Releases".

Pressurizer Pressure

In each of the pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power, the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10 ~~percent~~ of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10 ~~percent~~ of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approxi-m[illegible] 10 ~~percent~~ of RATED THERMAL POWER with a turbine impulse chamber pressure a[illegible]roximately 10 ~~percent~~ of full power equivalent); and on increasing power, a[illegible]atically reinstated by P-7.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10 ~~percent~~ % of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10 ~~percent~~ of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 35% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90 ~~percent~~ of nominal full loop flow. Conversely on decreasing power between P-8 and the P-7 an automatic reactor trip will occur on loss of flow in more than one loop and below P-7 the trip function is automatically blocked.

Overtemperature ΔT

The Overtemperature ~~delta~~ Δ T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 3.45 x 10^6 lbs/hour. The Steam Generator Low Water Level portion of the trip is activated when the water level drops below 25 ~~percent~~, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized. for Unit 1 and 1.47 x 10^6 lbs/hour for Unit 2.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide ~~reactor~~ core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.9 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.3 seconds. On decreasing power, the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10 ~~percent~~ of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10 ~~percent~~ of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine Trip initiates a Reactor trip. On decreasing power, the Turbine trip is automatically blocked by P-7 (a power level of approximately 10 ~~percent~~ % of RATED THERMAL POWER with a turbine impulse chamber at approximately 10 ~~percent~~ % of full power equivalent); and on increasing power, reinstated automatically by P-7.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide ~~reactor~~ core protection against DNB. The Open/Close Position trips assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. No credit was taken in the ~~accident~~ analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Trip System. Above P-7 (a power level of approximately 10 ~~percent~~ of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10 ~~percent~~ of full power equivalent) an automatic Reactor trip will occur if more than one reactor coolant pump breaker is opened. Above P-8 (a power level of approximately 35 ~~percent~~ of RATED THERMAL POWER) an automatic Reactor trip will occur if one reactor coolant pump breaker is opened. On decreasing power between P-8 and P-7 an automatic Reactor trip will occur if more than one reactor coolant pump breaker is opened and below P-7 the trip function is automatically blocked.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power, P-6 allows the manual block of the Source Range ~~reactor~~ trip and de-energizing of the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power, P-7 automatically enables Reactor trips on low flow in more than one ~~primary~~ reactor coolant loop, more than one reactor coolant pump breaker open, reactor coolant pump bus undervoltage and underfrequency, Turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

Reactor Trip System Interlocks (Continued)

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor ~~primary~~ coolant loops, and one or more reactor coolant pump breakers open. On decreasing power the P-8 automatically blocks the above listed trips.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range reactor trip and the Low Setpoint Power Range ~~reactor~~ trip; and automatically blocks the Source Range ~~reactor~~ trip and de-energizes the Source Range high voltage power. On decreasing power, the Intermediate Range ~~reactor~~ trip and the Low Setpoint Power Range ~~reactor~~ trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7

Seismic Trip

The Seismic trip is provided to automatically shutdown the reactor in the event of a seismic occurrence which corresponds in magnitude to the Double Design Earthquake. No credit was taken for operation of the Seismic trip in the safety analysis; however, its functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Trip System.

SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

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 a MODE in which the Specification does not apply by placing it, as applicable, in:

a. At least HOT STANDBY within the next 6 hours,
b. At least HOT SHUTDOWN within the following 6 hours, and
c. 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 to failure to meet the Limiting Condition for Operation. Exceptions of these requirements are stated in the individual Specifications.

This Specification is not applicable in MODES 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION statements. Exceptions to these requirements are stated in the individual Specifications.

Insert 11

3.0.5 Limiting Conditions for Operation including the associated ACTION requirements shall apply to each unit individually unless otherwise indicated as follows:

a. Whenever the Limiting Conditions for Operation refers to systems or components which are shared by both units, the ACTION requirements will apply to both units simultaneously. This will be indicated in the ACTION section;

b. Whenever the Limiting Conditions for Operation applies to only one unit, this will be identified in the APPLICABILITY section of the specification; and

c. Whenever certain portions of a specification contain operating parameters, Setpoints, etc., which are diferent for each unit, this will be identified in parentheses, footnotes or body of the requirement.

APPLICABILITY

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.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications. 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 have been performed within the stated surveillance interval or as otherwise specified.

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);

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
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5. (Continued)
~~4.0.5 (Continued)~~

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; and

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

4.0.6 Surveillance Requirements shall apply to each unit individually unless otherwise indicated as stated in Specification 3.0.5 for individual specifications or whenever certain portions of a specification contain surveillance parameters different for each unit, which will be identified in parentheses or footnotes.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} Greater Than 200°F

LIMITING CONDITIONS FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% ~~delta~~ Δ k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.6% ~~delta~~ Δ k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% ~~delta~~ Δ k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODES 1 or 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e, ~~e.~~ below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exception Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% ~~delta~~ Δ k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 ~~Effective Full Power Days~~ EFPD after each fuel loading.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} Less Than or Equal to 200°F

LIMITING CONDITIONS FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.~~0%~~ ~~delta~~ Δ k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.~~0%~~ ~~delta~~ Δ k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.~~0%~~ ~~delta~~ Δ k/k:

a. Within ~~one~~ 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITIONS FOR OPERATION

3.1.1.3 The ~~M~~oderator ~~T~~emperature ~~C~~oefficient (MTC) shall be:

a. Less positive than 0 $\Delta k/k/°F$ for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition; or

b. Less negative than -3.9×10^{-4} $\Delta k/k/°F$ for all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only#.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 $\Delta k/k/°F$ within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. ~~In lieu of any other report required by Specification 6.9.2.~~ A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

*With K_{eff} greater than or equal to 1.

#See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a, above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.0×10^{-4} $\Delta k/k/°F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.0×10^{-4} $\Delta k/k/°F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITIONS FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 541°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore (T_{avg}) to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 551°F, with the T_{avg} - T_{ref} Deviation Alarm not reset.

#With K_{eff} greater than or equal to 1.
*See Special Test Exception Specification 3.10.3.

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE with motor-operated valves required to change position and pumps required to operate for boron injection capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is greater than or equal to 145°F when a flow path from the boric acid tanks is used, and

b. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

LIMITING CONDITIONS FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and

b. The flow path from the refueling water storage tank via a charging pump to the ~~Reactor Coolant System.~~ RCS

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

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% ~~delta~~ k/k at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the flow path from the refueling water storage tank inoperable, restore the flow path 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.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is greater than or equal to 145°F,

b. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position,

c. At least once per 18 months by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal, and

#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 323°F.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 10 gpm to the ~~Reactor Coolant System~~. RCS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 At least the above required charging pump shall be demonstrated OPERABLE when tested pursuant to Specification 4.0.5. In addition, when the above required charging pump is a centrifugal charging pump, verify that, on recirculation flow, the centrifugal charging pump develops a differential pressure of greater than or equal to 2400 psid.

4.1.2.3.2 All centrifugal charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable* at least once per 12 hours, except when the reactor vessel head is removed, by verifying that the motor breaker D.C. control power is de-energized.

*An inoperable pump may be made OPERABLE for testing per Specification 4.0.5 provided the discharge of the pump has been isolated from the Reactor Coolant System by an isolation valve with power removed from the valve operator, or by a sealed closed manual isolation valve.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% ~~delta~~ k/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE when tested pursuant to Specification 4.0.5. In addition, when the above required charging pumps include a centrifugal charging pump(s), verify that, on recirculation flow, each required centrifugal charging pump(s) develops a differential pressure of greater than or equal to 2400 psid.

4.1.2.4.2 All centrifugal charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable* at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 323°F by verifying that the motor breaker D.C. control power is de-energized.

#A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 323°F.

*An inoperable pump may be made OPERABLE for testing per Specification 4.0.5 provided the discharge of the pump has been isolated from the Reactor Coolant System by an isolation valve with power removed from the valve operator, or by a sealed closed manual isolation valve.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System and at least one associated heat tracing channel with:

1) A minimum contained borated water volume of 835 gallons,

2) a boron concentration between 20,000 and 22,500 ppm ~~of boron~~, and

3) A minimum solution temperature of 145°F.

b. The Refueling Water Storage Tank (RWST) with:

1) A minimum contained borated water volume of 50,000 gallons,

2) A minimum boron concentration of 2000 ppm, and

3) A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside ambient air temperature is less than 35°F.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITIONS FOR OPERATION

3.1.2.6. Each of the following borated water source(s) shall be OPERABLE:

a. A Boric Acid Storage System and at least one associated heat tracing channel with:

1) A minimum contained borated water volume of 5106 gallons,

A boron concentration

2) Between 20,000 and 22,500 ppm ~~of boron~~, and

3) A minimum solution temperature of 145°F.

b. The Refueling Water Storage Tank (RWST) with:

1) A contained borated water volume of greater than or equal to [illegible] gallons,

A boron concentration

2) Between 2000 and 2200 ppm ~~boron~~, and

3) A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the Boric Acid Storage System inoperable, restore the ~~storage~~ system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% ~~delta~~ k/k at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST ~~refueling water storage tank~~ inoperable, restore the tank 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.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) Verifying the Boric Acid Storage System solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 35°F.

(4)

~~With one or more full-length rods trippable but inoperable due to causes other than addressed by ACTION a, above, POWER OPERATION may continue provided the inoperable rod(s) are positioned within ± 12 steps (indicated position) of their group demand position.~~

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length ~~(~~shutdown and control~~)~~ rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group ~~step counter~~ demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

KEEP THESE WORDS

b. ~~With more than one full-length rod inoperable or misaligned from the group (step counter) demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.~~

STET

c. With one full-length rod ~~inoperable due to causes other than addressed by ACTION a., above, or~~ misaligned from its group ~~step counter~~ demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the above alignment requirements, or

2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figure 3.1-1; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

   a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

   b) THE SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and ~~F~~ $F_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within ~~one~~ the next hour and within the ~~next~~ following 4 hours the ~~Power Range~~ Neutron Flux ~~High~~ High Trip Setpoints ~~are~~ reduced to less than or equal to 85% of RATED THERMAL POWER.

(4)

~~d. With more than one full length rod misaligned from the group demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.~~

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

TABLE 3.1-1 (3)

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss Of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITIONS FOR OPERATION

3.1.3.2 The ~~shutdown and control~~ Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod position indicator per bank inoperable either:

  1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

  2. Reduce THERMAL POWER TO less than 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

  1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

  2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3*#, 4*# and 5*#.

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE ~~by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months~~. by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel at least once per 18 months.

*With the Reactor Trip System breakers in the closed position.

#See Special Test Exemptions Specification 3.10.~~5~~4

ROD DROP TIME

LIMITING CONDITIONS FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within ~~one~~ hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in Control Banks A, B, C or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-1a for Unit 1 and Figure 3.1-1b for Unit 2

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, ~~either~~:

a. Restore the control banks to within the limits within ~~two~~ 2 hours, or

b. Reduce THERMAL POWER within ~~two~~ 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figures, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.

UNIT 1

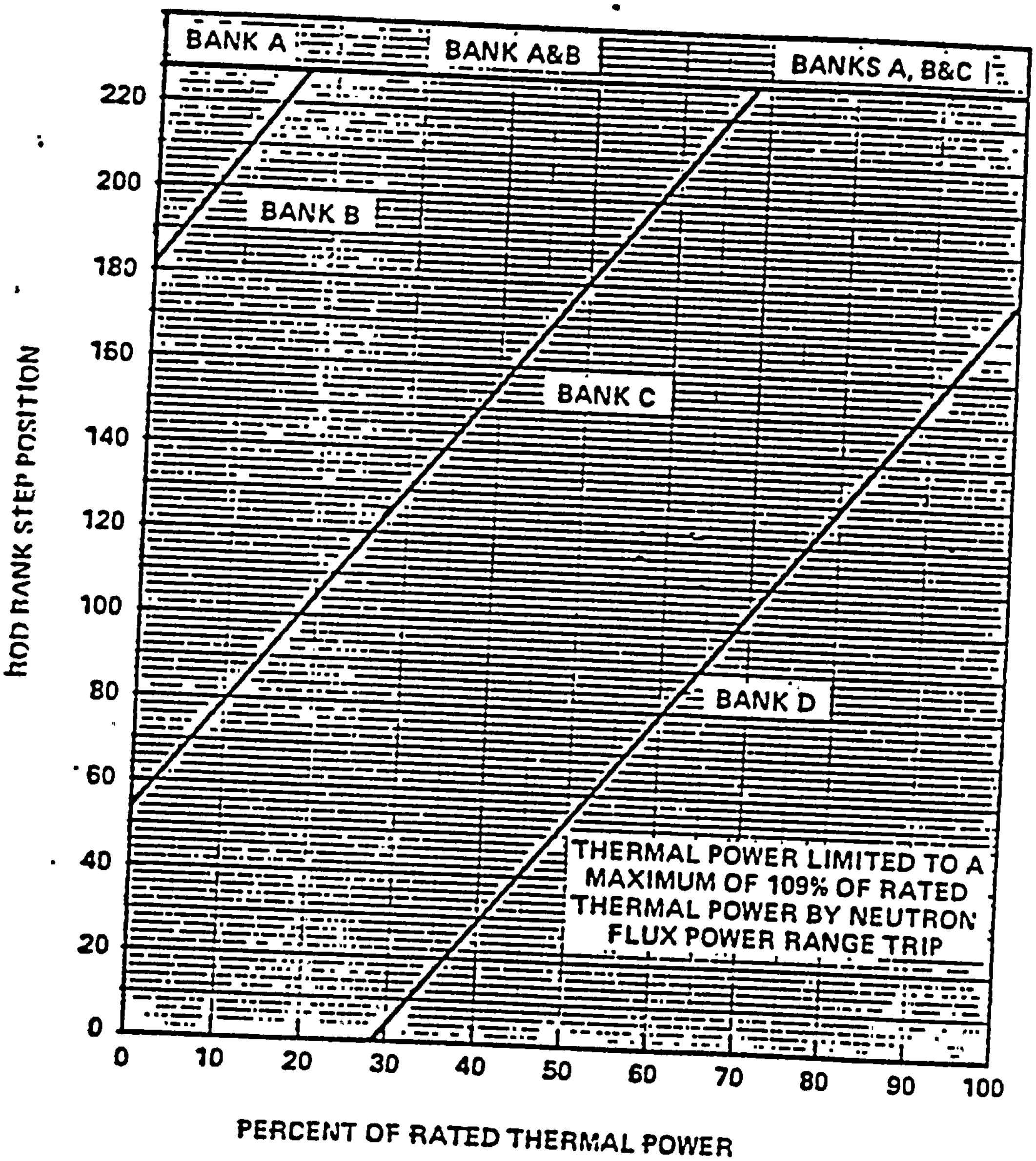

FIGURE 3.1-1a

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
100 STEP BANK OVERLAP

UNIT 2

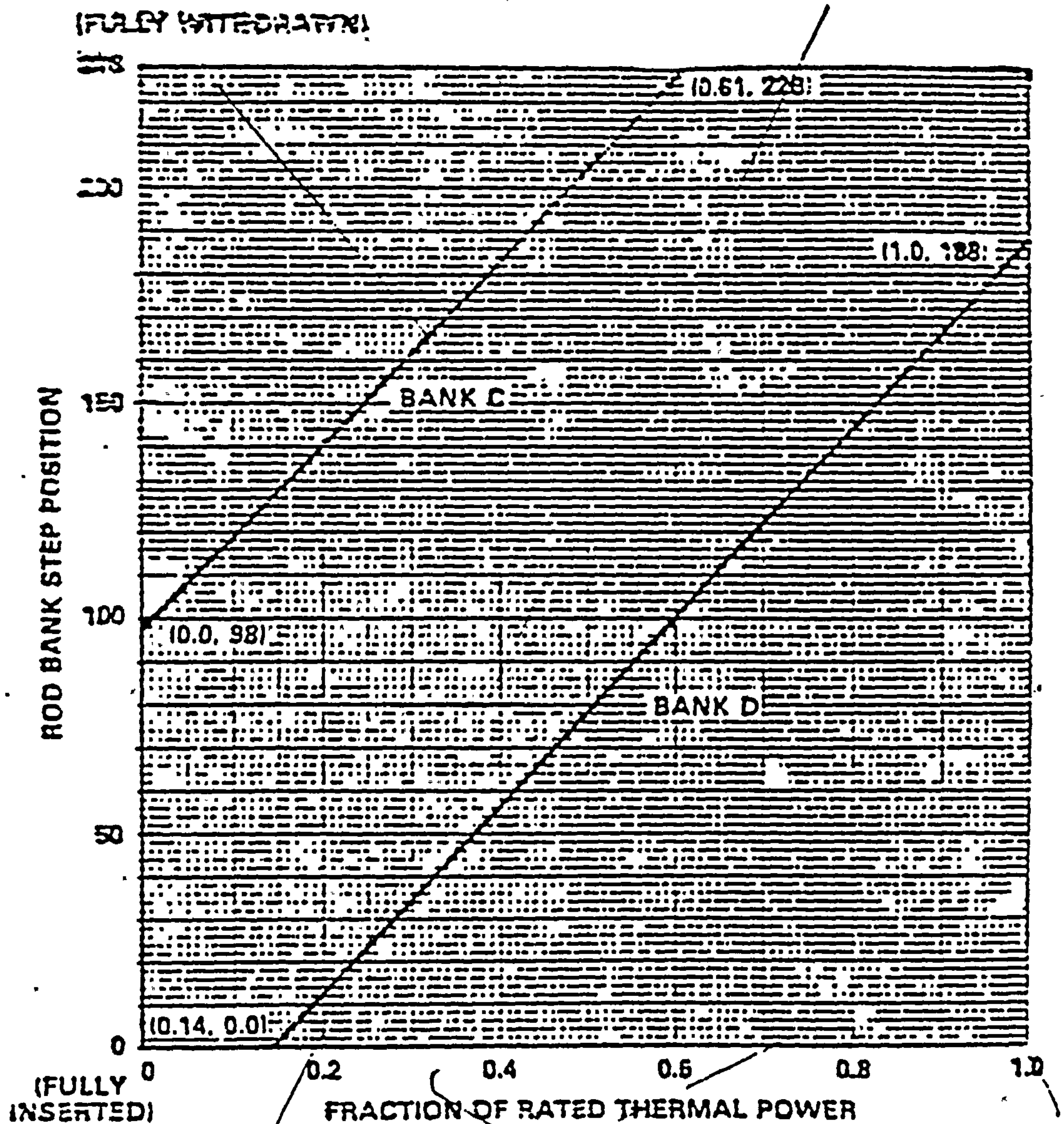

Figure 3.1-16
ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

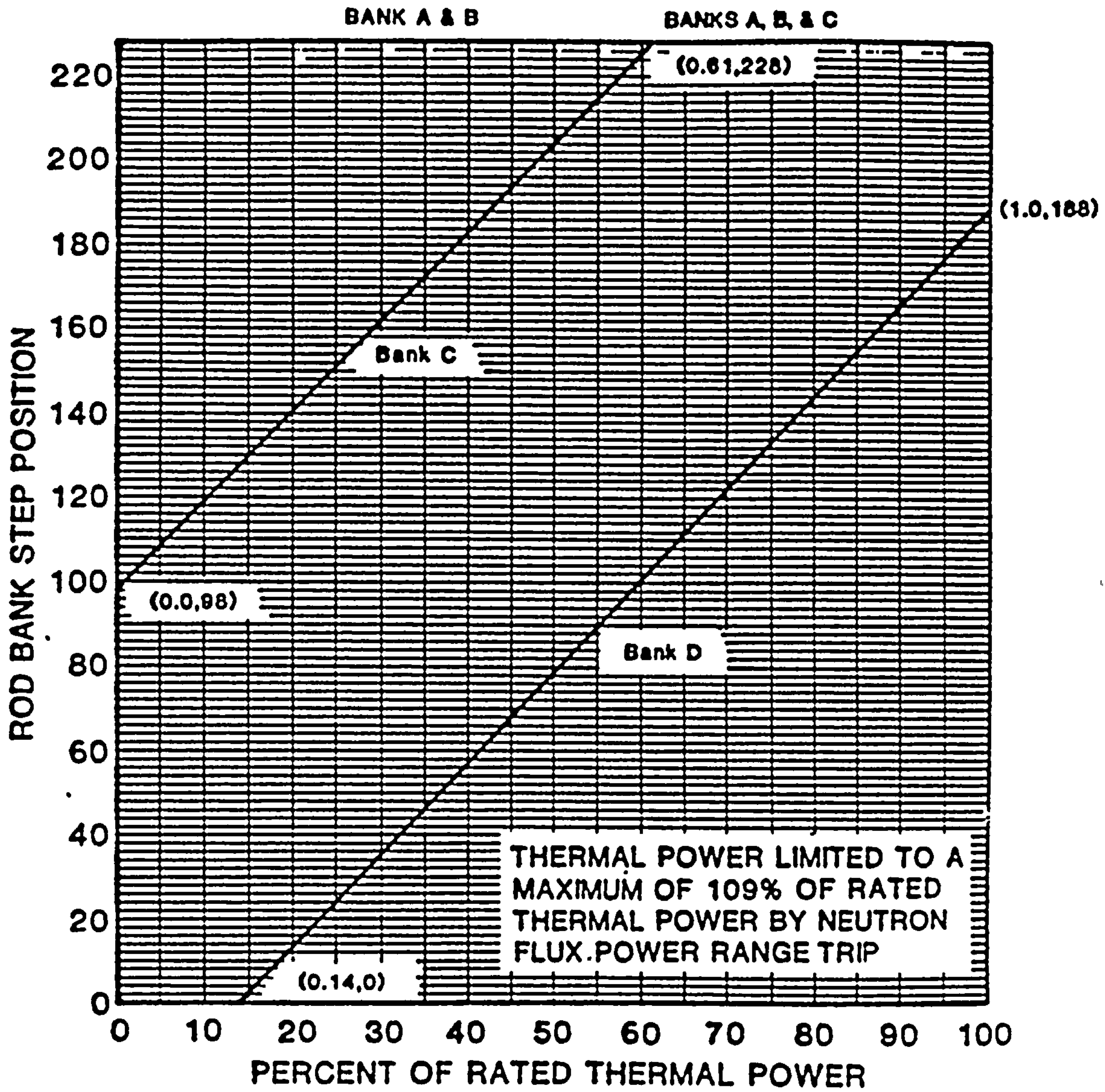

FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER FOR UNIT 2

-Insert for page 3/4 2-1-

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ±5% target band (flux difference units) about the target flux difference when THERMAL POWER is above 50% of RATED THERMAL POWER. The indicated AFD may deviate outside the above required target based at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1 above 15% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the ±5% target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes, either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux# - High# Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

*See Special Test Exception Specification 3.10.2.
#Surveillance testing of the Power Range Neutron Flux channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ±5% target band (flux difference units) about the target flux difference.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*.

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ±5% target band about the target flux difference and with THERMAL POWER:

1. Above 90% of RATED THERMAL POWER, within 15 minutes either:

   a) Restore the indicated AFD to within the target band limits, or

   b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

2. Between 50% and 90% of RATED THERMAL POWER:

   a) POWER OPERATION may continue provided:

      1) The indicated AFD has not been outside of the ±5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

      2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

   b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

3

*See Special Test Exceptions Specification 3.10.2.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION~~S~~ FOR OPERATION

ACTION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the ± 5% target band and ACTION a.2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ± 5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours. Power increases above 50% RATED THERMAL POWER do not require being within the target band provided the accumulative penalty deviation is not violated.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated ~~AXIAL FLUX DIFFERENCE~~ AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated ~~AXIAL FLUX DIFFERENCE~~ AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the ~~AXIAL FLUX DIFFERENCE~~ AFD Monitor Alarm is inoperable. The logged values of the indicated ~~AXIAL FLUX DIFFERENCE~~ AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ± 5% target band when ~~2~~ two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the ± 5% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each ~~one~~ minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each ~~one~~ minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and 0 ~~per cent~~ at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

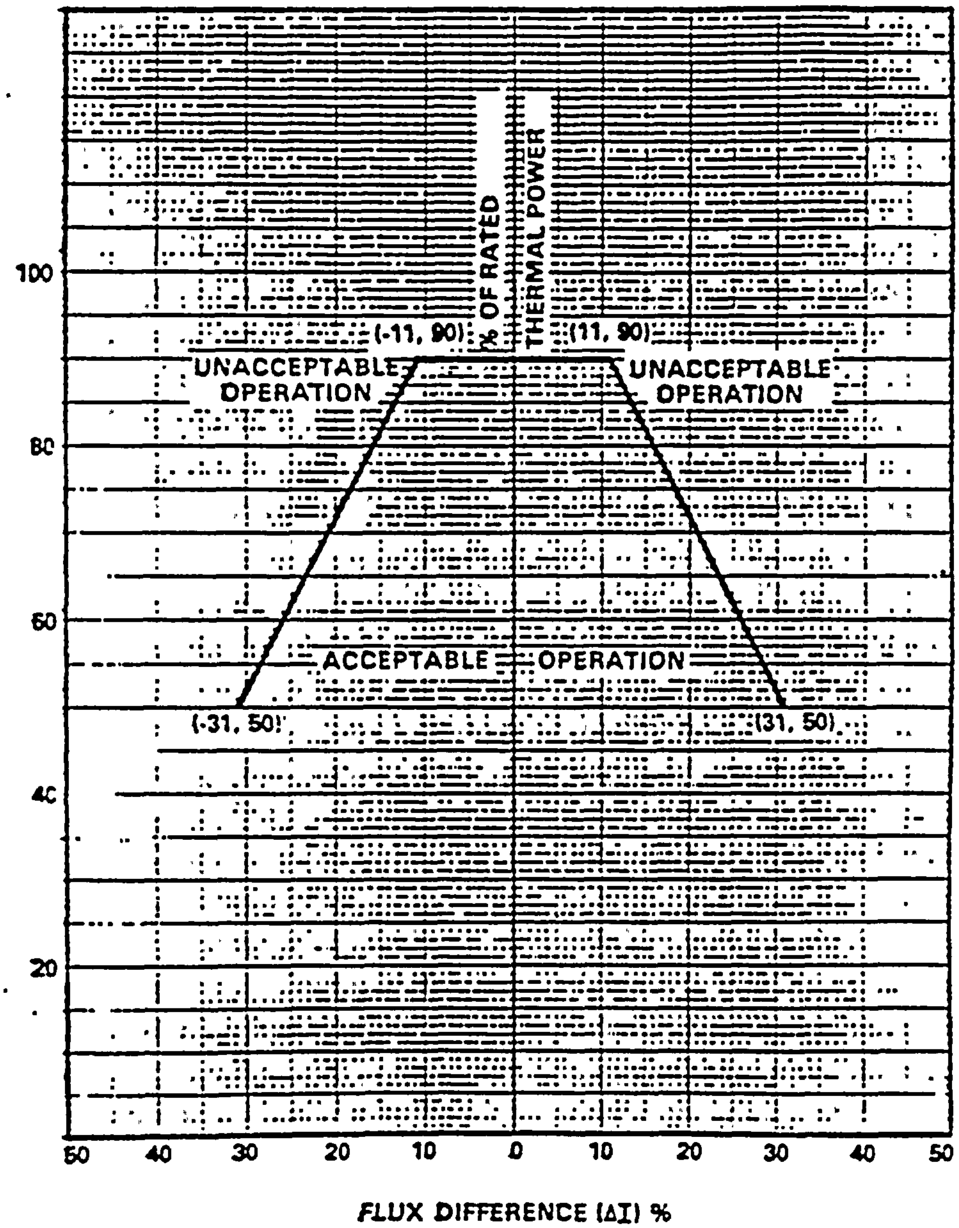

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS
AS A FUNCTION OF RATED THERMAL POWER

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.64] [K(Z)] \text{ for } P \leq 0.5$$

Where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

~~and~~ K(Z) ~~is~~ = the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-high Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ~~delta~~ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ~~delta~~ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER,

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in Specification 4.2.2.2.b. above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^{L} = F_{xy}^{RTP} [1+0.2(1-P)]$$

Where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1) When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^{L} either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 EFPD,

whichever occurs first.

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.~~12~~ 8.

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100% inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

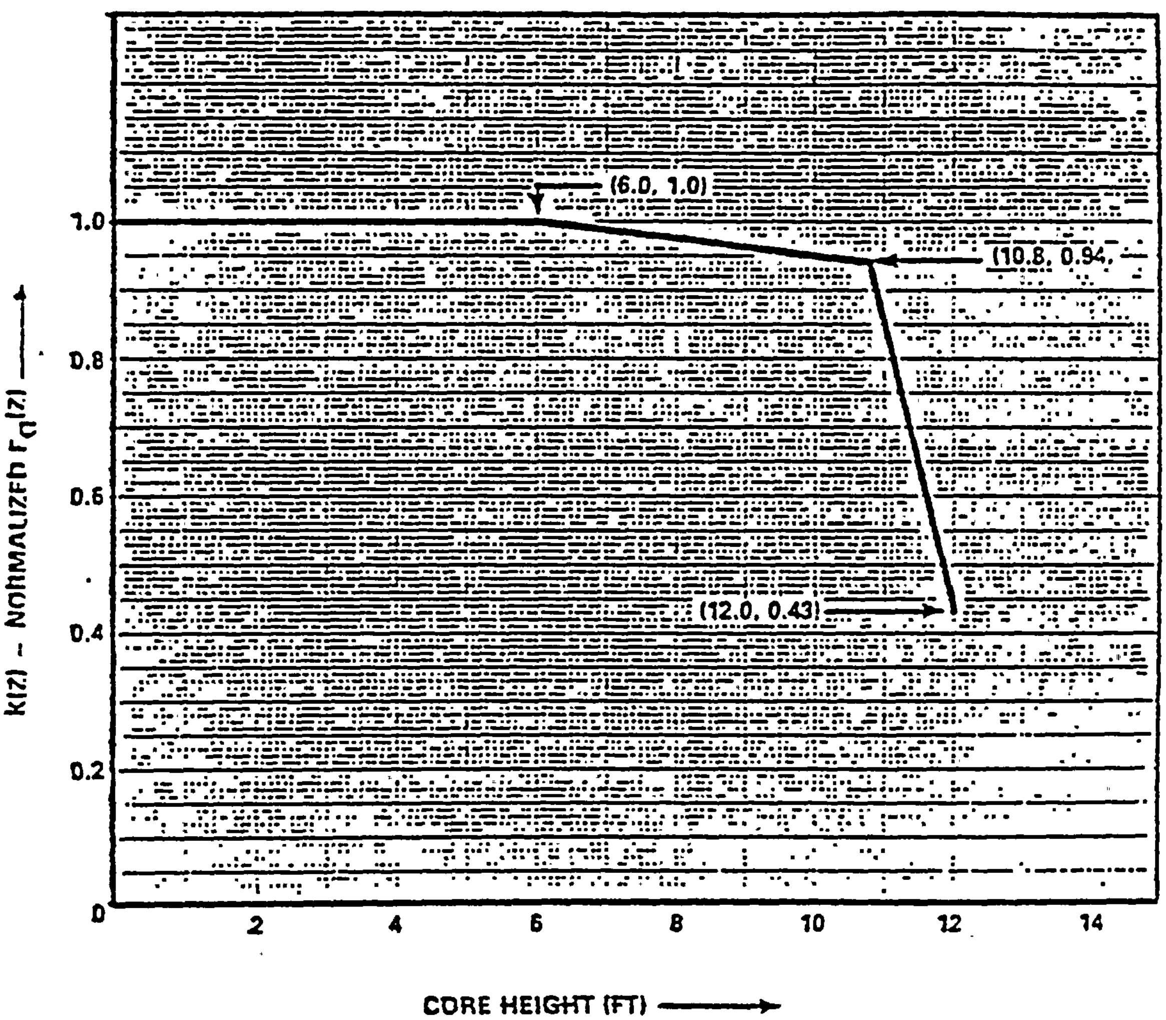

FIGURE 3.2-2

K(Z)- NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R, ~~R_2~~ shall be maintained within the region of allowable operation shown on Figure 3.2-3a for ~~4~~ four loop operation for Unit 1 and Figure 3.2-3b for Unit 2

(5)

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\,[1.0 + 0.2\,(1.0 - P)]}$,

~~b. $R_2 = \dfrac{R_1}{1 - RBP\,(BU)}$,~~

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$ and

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 include measurement uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$. ~~and~~

~~e. RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core).~~

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R, ~~R_2~~ outside the region of acceptable operation shown on Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R, ~~R_2~~ to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

POWER DISTRIBUTION LIMITS

LIMITING CONDITIONS FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R_1, R_2 and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that the combination of R_1, R_2 and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3a prior to exceeding the following THERMAL POWER levels: for Unit 1 and Figure 3.2-3b for Un

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R_1, R_2 shall be determined to be within the region of acceptable operation of Figure 3.2-3a for Unit 1 a. Figure 3.2-3b for Unit 2

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Uni at least once per 12 hours when the value~~s~~ of R_1 ~~and R_2~~, obtained per Specification 4.2.3.2, ~~are~~ is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per 18 months.

UNIT 1

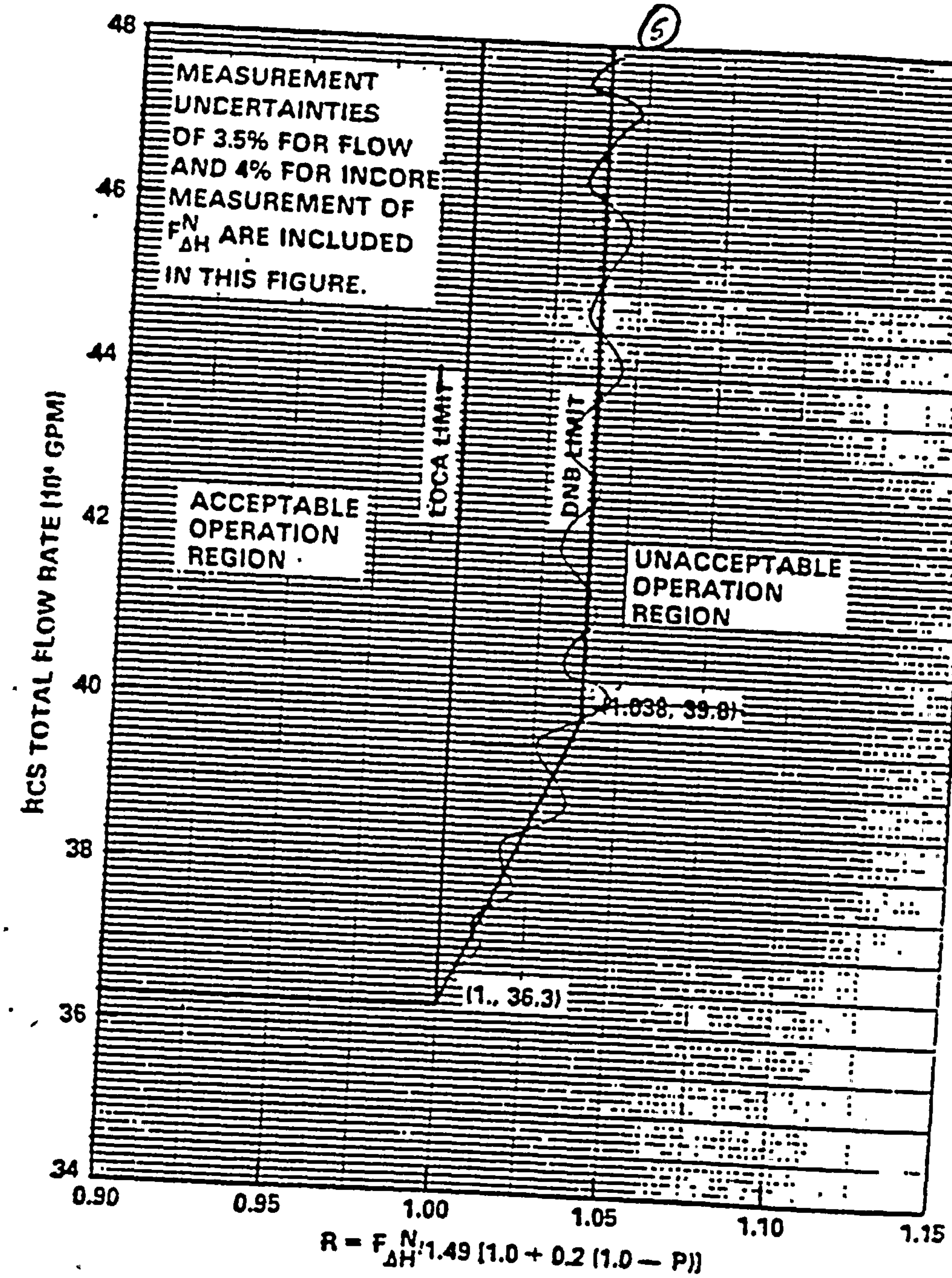

FIGURE 3.2-3a

RCS TOTAL FLOWRATE VERSUS R

UNIT 2

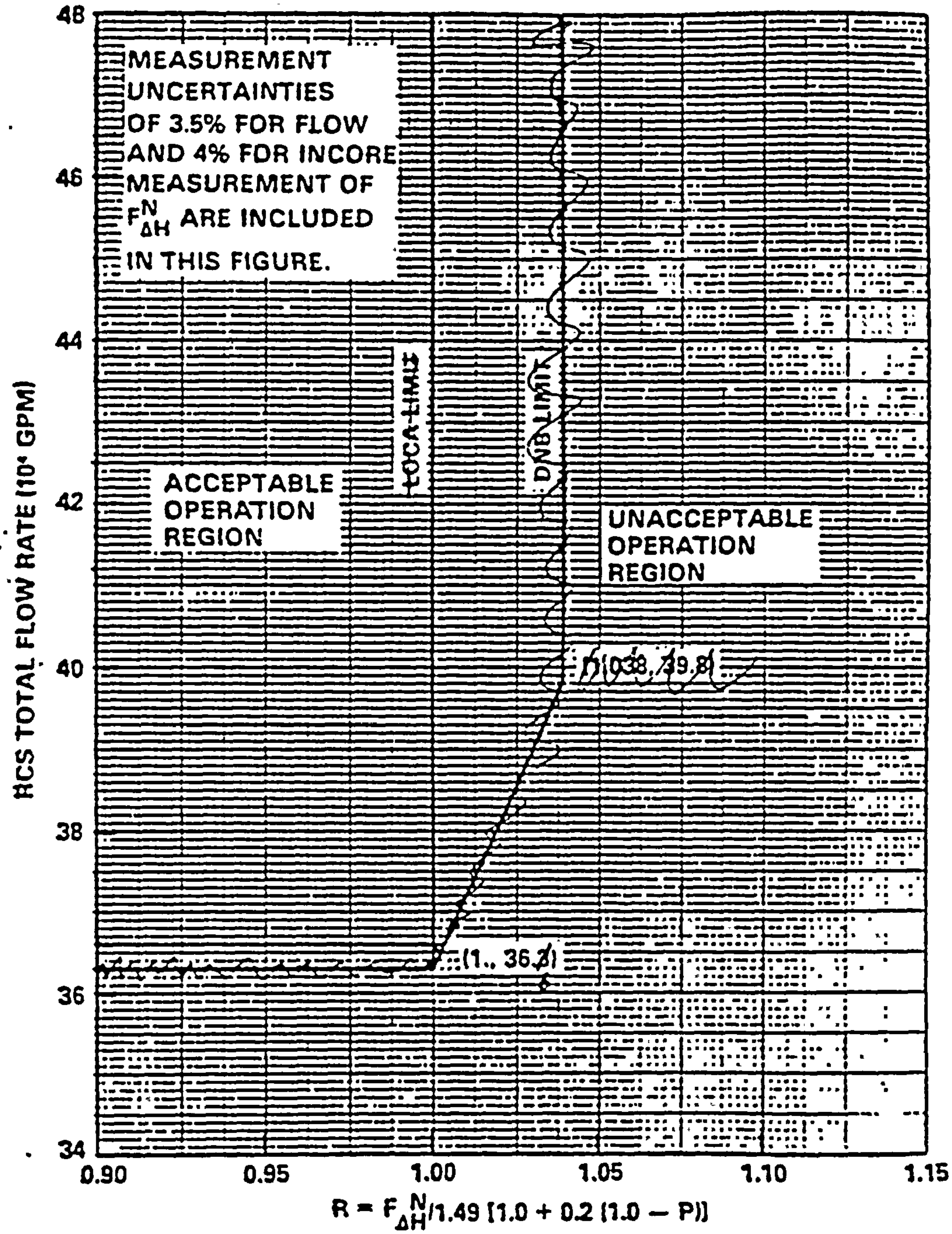

FIGURE 3.2-3b

RCS TOTAL FLOWRATE VERSUS R

[illegible]

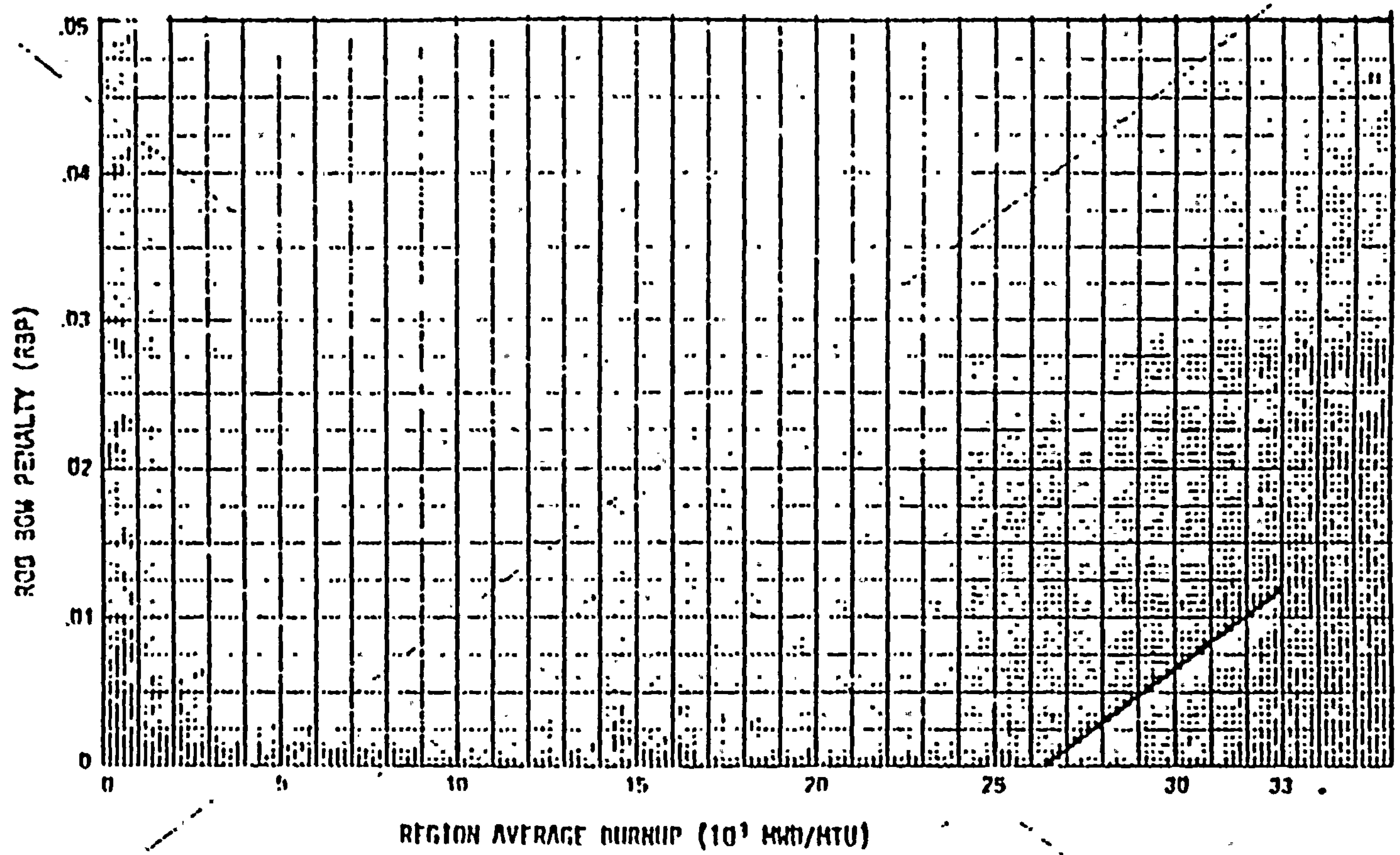

FIGURE 3.2-4

ROD BOW PENALTY AS A FUNCTION OF BURNUP

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Within 2 hours either:

      a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

   4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

*See Special Test Exceptions Specification 3.10.2.

ACTION (Continued)

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes;

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod, calculate the QUADRANT POWER TILT RATIO at least once per hour until THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER and:

1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours, and

2. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable to POWER OPERATION above 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 ~~percent~~ of RATED THERMAL POWER with one Power Range Channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from the 4 pairs of symmetric thimble locations, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS
ACTUAL Reactor Coolant System T_{avg}	[illegible]°F (UNIT 1) ≤ 582°F (UNIT 2)
ACTUAL Pressurizer Pressure	[illegible] 2220 psia* (UNIT 1) ≥ 2220 psia* (UNIT 2)

~~*Limit not applicable during either~~ a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

3/4.3 INSTRUMENTATION

3/4 3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	12
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	12
c. Shutdown	2	0	1	3, 4, and 5	5
7. Overtemperature ΔT	4	2	3	1, 2	6#
8. Overpower ΔT	4	2	3	1, 2	6#
9. Pressurizer Pressure-Low	4	2	3	1	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1	7#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12. Reactor Coolant Flow -Low					
a. Single Loop (Above P-8)	3/loop	2/loop in one loop	2/loop in each loop	1	7#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two loops	2/loop in each loop	1	7#
13. Steam Generator Water Level--Low-Low	3/S.G.	2/S.G. in one S.G.	2/S.G. in each S.G.	1, 2	7#
14. Steam Generator Water Level-Low Coincident With Steam/Feedwater Flow Mismatch	2 S.G. level and 2 stm./feed flow mismatch per S.G.	1 S.G. level coincident with 1 stm./feed flow mismatch in same S.G.	1 S.G. level and 2 stm./feed flow mismatch, or 2 S.G. level and 1 stm./feed flow mismatch per S.G.	1, 2	7#
15. Undervoltage-Reactor Coolant Pumps	2/bus	1/bus both busses	1/bus	1	6#
16. Underfrequency-Reactor Coolant Pumps	3/bus	2 on same bus	2/bus	1	6#
17. Turbine Trip					
a. Low Autostop Oil Pressure	3	2	2	1	7#
b. Turbine Stop Valve Closure	4	4	4	1	7#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Safety Injection Input from ESF	2	1	2	1, 2	11
19. Reactor Coolant Pump Breaker Position Trip					
a. Above P-8	1/breaker	1	1/breaker	1	9
b. Above P-7	1/breaker	2	1/breaker	1	10
20. Reactor Trip Breakers	2 2	1 1	2 2	1, 2 3*, 4*, 5*	11 12
21. Automatic Trip and Interlock Logic	2 2	1 1	2 2	1, 2 3*, 4*, 5*	11 12
22. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7 P-10 Input P-13 Input	4 2	2 1	3 2	1 1	8 8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	8
e. Turbine Impulse Chamber Pressure, P-13 (Input to P-7)	2	1	2	1	8
23. Seismic Trip	3 directions (x,y,z) in 3 locations	2/3 locations one direction	2/3 locations all directions	1, 2	6#

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

When
*~~With~~ the Reactor Trip System breakers in the closed position and the Control Rod Drive System capable of rod withdrawal.

#The provisions of Specification 3.0.4 are not applicable.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL and the Power Range, Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, or

b. Above the P-6 Setpoint, but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - 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:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within ~~one~~ 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With ~~one~~ a channel inoperable, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below the P-8 (Block of Low Reactor Coolant Pump Flow and Reactor Coolant Pump Breaker Position) Setpoint within the next 2 hours. Operation below the P-8 setpoint may continue pursuant to ACTION 10.

ACTION 10 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 11 - With the number of channels OPERABLE at least one less than the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 12 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	~~N.A.~~ ≤ 0.5 seconds*
7. Overtemperature ΔT	< 4 seconds*
8. Overpower ΔT	N.A.
9. Pressurizer Pressure--Low	< 2 seconds
10. Pressurizer Pressure--High	≤ 2 seconds
11. Pressurizer Water Level--High	N.A.

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Reactor Coolant Flow - Low	
a. Single Loop (Above P-8)	< 1 second
b. Two Loops (Above P-7 and below P-8)	< 1 second
13. Steam Generator Water Level--Low-Low	< 2 seconds
14. Steam Generator Water Level-Low Coincident With Steam/Feedwater Flow Mismatch	N.A.
15. Undervoltage-Reactor Coolant Pumps	< 1.2 seconds
16. Underfrequency-Reactor Coolant Pumps	< 0.6 second
17. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve	N.A.
18. Safety Injection Input from ESF	N.A.
19. Reactor Coolant Pump Breaker Position Trip	N.A.
20. Reactor Trip Breakers	N.A.
21. Automatic Trip and Interlock Logic	N.A.
22. Reactor Trip System Interlocks	N.A.
23. Seismic Trip	N.A.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	M	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	M	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1),M	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1),M(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R(12)	M	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	M	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	M	N.A.	N.A.	1
10. Pressurizer Pressure--High	S	R	M	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	M	N.A.	N.A.	1
12. Reactor Coolant Flow - Low	S	R	M	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	1, 2
14. Steam Generator Water Level - Low Coincident with Steam/Feedwater Flow Mismatch	S	R	M	N.A.	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
17. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	S/U(1, 10)	N.A.	1
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A	1, 2
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	R	N.A.	1
20. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	M	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	M(8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	M(8)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
20. Reactor Trip System Interlocks (Continued)						
d. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M(8)	N.A.	N.A.	1
21. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7,11)	N.A.	1, 2, 3*, 4*, 5*
22. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
23. Seismic Trip	N.A.	R	N.A.	SA	R	1, 2

TABLE NOTATIONS

* - When ~~With~~ the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

\## - Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

\### - Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) - Compare incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - Detector plateau curves shall be obtained and evaluated. For the Intermediate Range and Power Range Neutron Flux Channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) - Incore - Excore Calibration above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) - Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) - With power greater than or equal to the Interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) - Monthly Surveillance in MODES 3*, 4* and 5* shall also include verification that Permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(10) - Setpoint verification is not applicable.

(11) - At least once per 18 months and following maintenance or adjustment of the Reactor trip breaker, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include verification of the independence of the Undervoltage trip and Shunt trip.

(12) - CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation Channel or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the Trip Setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by the performance of the Engineered Safety Features Actuation System Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (REACTOR TRIP, FEEDWATER ISOLATION, ~~CONTROL ROOM ISOLATION,~~ START DIESEL GENERATORS, CONTAINMENT Fan ~~COOLING FANS~~ Cooler Units AND COMPONENT COOLING WATER) 3					
a. Manual Initiation	2	1	2	1, 2, 3, 4	19
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High	3	2	2	1, 2, 3,4	15*
d. Pressurizer Pressure - Low	4	2	3	1, 2, 3#	20*
e. Differential Pressure Between Steam Lines - High	3/steam line	2/steam line any steam line	2/steam line	1, 2, ~~3##~~ 3	15*
f. Steam Flow in Two Steam Lines-High	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, 3##	15*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
SAFETY INJECTION (REACTOR TRIP, FEEDWATER ISOLATION ~~CONTROL ROOM ISOLATION~~, START DIESEL GENERATORS CONTAINMENT ~~COOLING FANS AND~~ Fan Cooler Units COMPONENT COOLING WATER) (Continued))					
f. Coincident With Either T_{avg}--Low-Low	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 3 loops	1, 2, 3##	15*
Or, ~~Coincident With~~					
Steam Line Pressure-Low	1 pressure/loop	1 pressure any 2 loops	1 pressure any 3 loops	1, 2, 3##	15*
2. CONTAINMENT SPRAY					
a. Manual	2	2 with 2 coincident switches	2	1, 2, 3, 4	19
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-High	4	2	3	1, 2, 3	17
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	2	1	2	1, 2, 3, 4	19
2) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
CONTAINMENT ISOLATION (continued)					
2 3) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Phase "B" Isolation					
1) Manual	2	2 with 2 coincident switches	2	1, 2, 3, 4	19
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-High	4	2	3	1, 2, 3	17
c. Containment Ventilation Isolation					
1) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	18*
2) Plant Vent Noble Gas Activity-High RM-14A and 14B	2	1	2	1, 2, 3, 4	18*
3) Safety Injection	See item 1. above for all Safety Injection initiating functions and requirements.				

3)

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. STEAM LINE ISOLATION					
a. Manual	1 manual switch/steam line	1 manual switch/steam line	1 manual switch/operating steam line	1, 2, 3, 4	24
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
c. Containment Pressure--High-High	4	2	3	1, 2, 3	17
d. Steam Flow in Two Steam Lines--High	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, 3 ##	15*
Coincident With Either T_{avg}--Low-Low	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 3 loops	1, 2, 3 ##	15*
Or, ~~Coincident With~~					
Steam Line Pressure-Low	1 pressure/loop	1 pressure any 2 loops	1 pressure any 3 loops	1, 2, 3 ##	15*
5. TURBINE TRIP & FEEDWATER ISOLATION					
b. ~~a~~. Steam Generator Water Level--High-High	3/stm. gen.	2/stm. gen. in any operating stm. gen.	2/stm. gen. in each operating stm. gen.	1, 2	15*
a. ~~b~~. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	~~27~~ 25

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. ~~AUXILIARY FEEDWATER~~					
a. Manual Initiation	1 manual switch/pump	1 manual switch/pump	1 manual switch/pump	1, 2, 3	24
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
c. Stm. Gen. Water Level-Low-Low					
~~i.~~ 1) Start Motor-Driven Pumps	3/stm. gen.	2/stm. gen. in any operating stm. gen.	2/stm. gen. in each operating stm. gen.	1, 2, 3	15*
~~ii.~~ 2) Start Turbine-Driven Pump	3/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	2/stm. gen in each operating stm. gen.	1, 2, 3	15*
d. Undervoltage-RCP Bus Start Turbine-Driven Pump	2/bus	1/bus on both busses	1/bus	1	20*
e. Safety Injection Start Motor-Driven Pumps	See item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. LOSS OF POWER (4.16 KV Emergency Bus Undervoltage)					
a. First Level				1, 2, 3, 4	
1) Diesel Start	1/Bus	1/Bus	1/Bus		16
2) Initiation of Load Shed	2/Bus	2/Bus	2/Bus		16
b. Second Level				1, 2, 3, 4	
1) Undervoltage Relays	2/Bus	2/Bus	2/Bus		16
2) Timers to Start Diesel	1/Bus	1/Bus	1/Bus		16
3) Timers to Shed Load	1/Bus	1/Bus	1/Bus		16
8. ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INTERLOCKS					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	21
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	21
c. Reactor Trip, P-4	2	2	2	1, 2, 3	23

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

##Trip function may be blocked in this MODE below the P-12 (Low-Low Tavg Interlock) Setpoint.

*The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, declare the affected Emergency Diesel Generator(s) inoperable and comply with the ACTION statements of Specification 3.8.1.1; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 18 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves (RCV-11, 12, FCV 660, 661, 662, 663, 664) are maintained closed.

5

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 19 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 20 - 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:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirements is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 21 - With less than the Minimum Number of Channels OPERABLE, within 1 ~~one~~ hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 22 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 24 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated pump or valve inoperable and take the ACTION required by Specification 3.7.1.5 or 3.7.1.2 as applicable.

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (REACTOR TRIP, FEEDWATER ISOLATION, ~~CONTROL ROOM ISOLATION~~, START DIESEL GENERATORS, CONTAINMENT ~~COOLING~~ Fan Cooler Units ~~FANS~~ AND COMPONENT COOLING WATER)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High	≤ 3 psig	≤ 3.5 psig
d. Pressurizer Pressure--Low	≥ 1850 psig	≥ 1840 psig
e. Differential Pressure Between Steam Lines--High	≤ 100 psi	≤ 112 psi
f. Steam Flow in Two Steam Lines--High	≤ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load.	≤ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load.
Coincident With Either		
1) T_{avg}--Low-Low, or	≥ 543°F	≥ 541°F
2) Steam Line Pressure--Low	≥ 600 psig	≥ ~~585~~ 580 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. CONTAINMENT SPRAY		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	$\leq$ 22 psig	$\leq$ 24 psig
3. CONTAINMENT ISOLATION		
a. Phase "A" Isolation		
1) Manual	N.A.	N.A.
3 ~~2~~) Safety Injection	See item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
2 ~~3~~) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Phase "B" Isolation		
1) Manual	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure--High-High	$\leq$ 22 psig	$\leq$ 24 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. ~~CONTAINMENT ISOLATION~~ (continued)		
c. Containment Ventilation Isolation		
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
2) Plant Vent Noble Gas Activity-High	Per Specification 3.3.3.10	
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
4. ~~STEAM LINE ISOLATION~~		
a. Manual	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High	≤ 22 psig	≤ 24 psig
d. Steam Flow in Two Steam Lines--High	< A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load.	< A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
STEAM LINE ISOLATION (continued)		
Coincident With Either		
1) T_{avg}--Low-Low, or	$\geq$ 543°F	$\geq$ 541°F
2) Steam Line Pressure--Low	$\geq$ 600 psig	$\geq$ ~~585~~ 590 psig (2)
5. TURBINE TRIP AND FEED WATER ISOLATION		
b a. Steam Generator Water Level--High-High	< 67% of narrow range instrument span each steam generator	< 68% of narrow range instrument span each steam generator
a b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
6. AUXILIARY FEEDWATER		
a. Manual	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level-Low-Low	> 15% of narrow range instrument span each steam generator.	> 14% of narrow range instrument span each steam generator.
d. Undervoltage - RCP	$\geq$ 8050 volts	$\geq$ 7935 volts
e. Safety Injection	See item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. LOSS OF POWER (4.16 kv Emergency Bus Undervoltage)		
a. First Level		
1) Diesel Start	≥ 0 volts with a ≤ 0.8 second time delay and ≥ 2583 volts with a ≤ 10.0 second time delay	≥ 0 volts with a ≤ 0.8 second time delay and ≥ 2583 volts with a ≤ 10.0 second time delay
2) Initiation of Load Shed	One relay ≥ 0 volts with a ≤ 4.0 second time delay and ≥ 2583 volts with a ≤ 25.0 second time delay with one relay ≥ 2870 volts, instantaneous	One relay ≥ 0 volts with a ≤ 4.0 second time delay and ≥ 2583 volts with a ≤ 25.0 second time delay with one relay ≥ 2870 volts, instantaneous
b. Second Level		
1) Diesel Start	≥ 3600 volts with a ≤ 10.0 second time delay	≥ 3600 volts with a ≤ 10.0 second time delay
2) Initiation of Load Shed	≥ 3600 volts with a ≤ 20.0 second time delay	≥ 3600 volts with a ≤ 20.0 second time delay
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
a. Pressurizer Pressure, P-11	≤ 1915 psig	≤ 1925 psig
b. Low-Low T_{avg}, P-12 increasing	543°F	≤ 545°F
decreasing	543°F	≥ 541°F
c. Reactor Trip, P-4	N.A.	N.A.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
1) Feedwater Isolation	N.A.
2) Reactor Trip ~~(SI)~~	N.A.
3) ~~Containment Isolation~~ Phase "A" Isolation	N.A.
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater ~~Pumps~~	N.A.
6) Component Cooling Water ~~Pumps~~	N.A.
7) Containment Fan Cooler ~~Units~~	N.A.
8) Auxiliary Salt Water Pumps	N.A.
~~b. Containment Spray~~	N.A.
~~c. Containment Isolation Phase "B" Isolation~~	N.A.
~~Containment Ventilation Isolation~~	N.A.
~~d. e. Containment Isolation-Phase "A" Isolation~~	N.A.
~~Containment Ventilation Isolation~~	N.A.
STET ~~e.~~ d. Steam Line Isolation	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ 27~~.0~~[1]
~~b.~~ 1) Reactor Trip ~~(from SI)~~	≤ 2~~.0~~
~~c.~~ 2) Feedwater Isolation	≤ 63~~.0~~[2]
~~d.~~ 3) ~~Containment Isolation~~ Phase "A" Isolation	≤ 18~~.0~~[4]/28~~.0~~[5]
~~e.~~ 4) Containment Ventilation Isolation	N.A.
~~f.~~ 5) Auxiliary Feedwater ~~Pumps~~	≤ 60~~.0~~[1]
~~g.~~ 6) Component Cooling Water ~~Pumps~~	≤ 38~~.0~~[4]/48~~.0~~[5]
~~h.~~ 7) Containment Fan Cooler ~~Units~~	≤ 40[1]
~~i.~~ 8) Auxiliary Salt Water Pumps	≤ 48~~.0~~[4]/58~~.0~~[5]
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27~~.0~~[1]/12~~.0~~[4]
~~b.~~ 1) Reactor Trip ~~(from SI)~~	≤ 2~~.0~~
~~c.~~ 2) Feedwater Isolation	≤ 63~~.0~~[2]
~~d.~~ 3) ~~Containment Isolation~~ Phase "A" Isolation	≤ 18~~.0~~[4]
~~e.~~ 4) Containment Ventilation Isolation	N.A.
~~f.~~ 5) Auxiliary Feedwater ~~Pumps~~	≤ 60~~.0~~[1]
~~g.~~ 6) Component Cooling Water ~~Pumps~~	≤ 48~~.0~~[1]/38~~.0~~[4]
~~h.~~ 7) Containment Fan Cooler ~~Units~~	≤ 40[1]
~~i.~~ 8) Auxiliary Saltwater Pumps	≤ 58~~.0~~[1]/48~~.0~~[4]

b. Phase "B" Isolation
1) Containment spray (Coincident with SI signal)
2) Containment Ventilation Isolation
c. Phase "A" Isolation
1) Containment Ventilation Isolation

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	≤ 13~~.0~~(4)/23~~.0~~(5)
b. Reactor Trip ~~(from SI)~~	≤ 2~~.0~~
c. Feedwater Isolation	≤ 63~~.0~~(2)
d. ~~Containment Isolation~~ Phase "A" Isolation	≤ 18~~.0~~(4)/28~~.0~~(5)
e. Containment Ventilation Isolation	N.A.
f. Auxiliary Feedwater ~~Pumps~~	≤ 60~~.0~~(1)
g. Component Cooling Water ~~Pumps~~	≤ 38~~.0~~(4)/48~~.0~~(5)
h. Containment Fan Cooler Units	≤ 40(1)
i. Auxiliary Saltwater Pumps	≤ 48~~.0~~(4)/58~~.0~~(5)
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	≤ 15~~.0~~(4)/25~~.0~~(5)
b. Reactor Trip ~~(from SI)~~	≤ 4~~.0~~
c. Feedwater Isolation	≤ 65~~.0~~(2)
d. ~~Containment Isolation~~ Phase "A" Isolation	≤ 20~~.0~~(4)/30~~.0~~(5)
e. Containment Ventilation Isolation	N.A.
f. Auxiliary Feedwater ~~Pumps~~	≤ 60~~.0~~(1)
g. Component Cooling Water ~~Pumps~~	≤ 40.0(4)/50.0(5)
h. Steam Line Isolation	≤ 10~~.0~~
i. Containment Fan Cooler Units	≤ 40(1)
j. Auxiliary Saltwater Pumps	≤ 50~~.0~~(4)/60~~.0~~(5)
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 13~~.0~~(4)/23~~.0~~(5)
b. Reactor Trip ~~(from SI)~~	≤ 2~~.0~~
c. Feedwater Isolation	≤ 63.0(2)
d. ~~Containment Isolation~~ Phase "A" Isolation	≤ 18~~.0~~(4)/28~~.0~~(5)
e. Containment Ventilation Isolation	N.A.
f. Auxiliary Feedwater ~~Pumps~~	≤ 60~~.0~~(1)
g. Component Cooling Water ~~Pumps~~	≤ 38~~.0~~(4)/48~~.0~~(5)
h. Steam Line Isolation	≤ 8.0
i. Containment Fan Cooler Units	≤ 40(1)
j. Auxiliary Saltwater Pumps	≤ 48~~.0~~(4)/58~~.0~~(5)
7. Containment Pressure--High-High	
a. Containment Spray	≤ 48.5(6)
b. ~~Containment Isolation~~ Phase "B" Isolation	N.A.
c. Steam Line Isolation	≤ 7~~.0~~

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
8. Steam Generator Water Level--High-High	
a. Turbine Trip ~~Reactor Trip~~	≤ 2.5
b. Feedwater Isolation	≤ 66.~~0~~(2)
9. Steam Generator Water Level Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Turbine-Driven Auxiliary Feedwater Pump	≤ 60
10. RCP Bus Undervoltage	
Turbine-Driven Auxiliary Feedwater Pump	≤ 60
11. Plant Vent Noble Gas Activity-High	
Containment Ventilation Isolation	≤ 11

TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps (where applicable).

(2) Feedwater System overall response time shall include verification of each individual Feedwater System valve closure time as shown below:

Valve	Closure Time (not including instrumentation delays)
FCV-43[illegible]	≤ 60 seconds
43[illegible]	≤ 60 seconds
44[illegible]	≤ 60 seconds
44[illegible]	≤ 60 seconds
5[illegible]	≤ 5 seconds
5[illegible]	≤ 5 seconds
53[illegible]	≤ 5 seconds
54[illegible]	≤ 5 seconds

(3) Diesel generator starting and loading delays included.

(4) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps (where applicable).

(5) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(6) The maximum response time of 48.5 seconds is the time from when the containment pressure exceeds the High-High Setpoint until the spray pump is started and the discharge valve travels to the fully open position assuming off-site power is not available. The time of 48.5 seconds includes the 28-second maximum delay related to ESF loading sequence. Spray riser piping fill time is not included. The 80-second maximum spray delay time does not include the time from LOCA start to "P" signal.

1510	≤ 5 seconds
1520	≤ 5 seconds
1530	≤ 5 seconds
1540	≤ 5 seconds

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TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION (REACTOR TRIP, FEEDWATER ISOLATION, ~~CONTROL ROOM ISOLATION~~ START DIESEL GENERATORS, CONTAINMENT ~~COOLING~~ FAN COOLER UNITS ~~FANS~~ AND COMPONENT COOLING WATER) (3)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
C.K. Containment Pressure-High	S	R	M	N.A.	N.A	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure--Low	S	R	M	N.A	N.A.	N.A.	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Flow in Two Steam Lines--High Coincident With Either	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
1) T_{avg}--Low-Low, or	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2) Steam Line Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2. CONTAINMENT SPRAY								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
3. CONTAINMENT ISOLATION								
a. Phase "A" Isolation								
1) Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
3 ~~2)~~ Safety Injection	See item 1. above for all Safety Injection Surveillance Requirements							
2 ~~3)~~ Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. Phase "B" Isolation								
1) Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
c. Containment Ventilation Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
2) Plant Vent Activity-High	S	R	M (2) (3)	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Injection Surveillance Requirements.							
4. STEAM LINE ISOLATION								
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident With Either	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
1) T_{avg}--Low-Low or	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2) Steam Line Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION								
a. Steam Generator Water Level--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
k. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
6. AUXILIARY FEEDWATER								
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Undervoltage - RCP	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1
e. Safety Injection	See item 1. above for all Safety Injection Surveillance Requirements.							
7. LOSS OF POWER								
a. 4.16 kV Emergency Bus Level 1	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4.16 kV Emergency Bus Level 2	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low, Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) For the Plant vent Activity - High Monitor, Only A CHANNEL FUNCTIONAL TEST shall be performed at least once every 31 days.

3

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING ~~INSTRUMENTATION~~ FOR PLANT OPERATION

LIMITING CONDITIONS FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-6 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
~~1. AREA MONITORS~~				
1. ~~a~~. Fuel Storage Area				
a. ~~1~~) Spent Fuel Pool	1	*	≤ 15 mR/hr	30 & 32**
b. ~~2~~). New Fuel Storage	1	*	≤ 15 mR/hr	30 & 32**
2. ~~b~~. Control Room 5 Ventilation ~~Isolation~~ Mode Change	~~1~~ 2***	All ~~MODES~~	≤ 2 mR/hr	34
~~2. PROCESS MONITORS~~				
3. ~~a~~. Containment				
a. ~~1~~) Gaseous Activity				
1 ~~a~~) Containment Ventilation Isolation (RM-14A or 14B)	1	6	Per Specification 3.3.3.10	33
2 ~~b~~) RCS Leakage	1	1, 2, 3, 4	N.A.	31
~~2)~~ b. Particulate Activity				
a) RCS Leakage	1	1, 2, 3, 4	N.A.	31

*With fuel in the spent fuel pool or new fuel storage vault.

**With irradiated fuel in the spent fuel pool.

*** One channel for each normal intake to the Control Room Ventilation System (common to both units).

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 30 - ~~With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.~~

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room ventilation system in a recirculation mode with the HEPA filter and charcoal adsorber system in operation.

With less than the Minimum Channels OPERABLE requirement operation may continue for up to 30 days provided an appropriate portable continuous monitor with the Same Alarm Setpoint or an individual qualified in Radiation Protection Procedures with a radiation dose rate monitoring device is provided in the fuel storage pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel storage pool area.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION for plant operations SURVEILLANCE REQUIREMENTS

	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
~~1. AREA MONITORS~~				
~~a.~~ 1. Fuel Storage Area				
~~1)~~ a. Spent Fuel Pool	S	R	M	*
~~2)~~ b. New Fuel Storage	S	R	M	*
~~b.~~ 2. Control Room Ventilation ~~Isolation~~ Mode Change	S	R	M	All ~~MODES~~
~~2. PROCESS MONITORS~~				
~~a.~~ 3. Containment				
~~1)~~ a. Gaseous Activity				
1 ~~a)~~ Containment Ventilation Isolation (RM-14A or 14B)	S	R	M	6
2 ~~b)~~ RCS Leakage ~~Detection~~	S	R	M	1, 2, 3, 4
~~2)~~ b. Particulate Activity				
~~a)~~ RCS Leakage ~~Detection~~	S	R	M	1, 2, 3, 4

*With fuel in the spent fuel pool or new fuel storage vault.

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITIONS FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}.

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}.

INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION~~S~~ FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation # shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION, as applicable, performed within 10 ~~5~~ days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within ~~10~~ 14 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

(7)

The Seismic Monitoring instrumentation is common to both units but located in Unit 1 or common areas.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelographs		
a. Containment Base Slab, El 85, 180°	± 1g	1*
b. Top Unit 1 Containment, El 303.5, 225°	± 2g	1
c. Aux Building, El 64	± 1g	1
2. Triaxial Peak Accelographs		
a. Containment Base Slab, El 89, 180°	± 2g	1
b. Top Unit 1 Containment, El 303.5, 225°	± 5g	1
c. Intake near ASW Pump 1-2 Bay, El 2	± 2g	1
d. Turbine Building, El 85, Machine Shop	± 2g	1
e. Aux Building, El 140, Hot Shop	± 2g	1
f. Aux Building, El 140, Near Control Room Door	± 2g	1
3. Triaxial Response-Spectrum Recorders		
a. Containment Base Slab, El 89, 180°	1.6 - 90 g 2-25.4 Hz	1

* With reactor control room indications or annunciation.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelographs			
a. Containment Base Slab, EL 89, 180°**	M*	R	SA
b. Top Unit 1 Containment, EL 303.5, 225°	M*	R	SA
c. Aux Building, EL 64	M*	R	SA
2. Triaxial Peak Accelographs			
a. Containment Base Slab, EL 89, 180°	N.A.	N.A.	N.A.
b. Top Unit 1 Containment, EL 303.5, 225°	N.A.	N.A.	N.A.
c. Intake near ASW Pump 1-2 Bay, EL 2	N.A.	N.A.	N.A.
d. Turbine Building, EL 85, Machine Shop	N.A.	N.A.	N.A.
e. Aux Building, EL 140, Hot Shop	N.A.	N.A.	N.A.
f. Aux Building, EL 140, Near Control Room door	N.A.	N.A.	N.A.
3. Triaxial Response-Spectrum Recorders			
a. Containment Base Slab, EL 89, 180°	N.A.	N.A.	N.A.

* Except seismic trigger.
** With reactor control room indications or annunciation.

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels# shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-5.

The meteorological monitoring instrumentation channels are common to both units.

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

METEOROLOGICAL TOWER INSTRUMENT AND LOCATION	MINIMUM OPERABLE
1. WIND SPEED	
a. Nominal Elev. 25'	1 of 2
b. Nominal Elev. 250'	
2. WIND DIRECTION	
a. Nominal Elev. 25'	1 of 2
b. Nominal Elev. 250'	
3. AIR TEMPERATURE - ~~DELTA~~ Δ T	
a. Nominal Elev. 250'-25'	1 of 2
b. Nominal Elev. 150'-25'	

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 25'	D	SA
b. Nominal Elev. 250'	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 25'	D	SA
b. Nominal Elev. 250'	D	SA
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 250'-25'	D	SA
b. Nominal Elev. 150'-25'	D	SA

INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the minimum channels required in Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-6.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	2 ~~MEASUREMENT RANGE~~	MINIMUM CHANNELS OPERABLE
1 1. Reactor Trip Breaker Indication	Reactor Trip Breaker	~~Open/Close~~	1/trip breaker
2. Pressurizer Pressure	Hot Shutdown Panel	~~1250 - 2500 psig~~	1
3. Pressurizer Level	Hot Shutdown Panel	~~0 - 100%~~	1
4. Steam Generator Pressure	Hot Shutdown Panel	~~0 - 1200 psig~~	1/steam generator
5. Steam Generator Wide Range Water Level	Hot Shutdown Panel	~~0 - 100%~~	1/steam generator
6. Condensate Storage Tank Water Level	Hot Shutdown Panel	~~0 - 100%~~ 3	1
7. Auxiliary Feedwater Flow	Hot Shutdown Panel	~~0 - 300 gpm~~	1/steam generator
8. Emergency Borate Flow	Hot Shutdown Panel	~~0 - 150 gpm~~	1
9. Charging Flow	Hot Shutdown Panel	~~0 - 200 gpm~~	1

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	N.A.	N.A.
2. Pressurizer Pressure	M	R
3. Pressurizer Level	M	R
4. Steam Generator Wide Range water Level	M	R
5. Steam Generator Pressure	M	R
6. Condensate Storage Tank water level	M	R
7. Auxiliary Feedwater Flow	M	R
8. Emergency Borate Flow	M	R
9. Charging Flow	M	R

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION: except for Reactor Vessel Level Indication System provided under

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours. (3)

b. With the number of OPERABLE accident monitoring instrumentation channels except the containment recirculation sump level-narrow range, the main steam line radiation monitor, the containment area radiation monitor-high range, and the plant vent radiation monitor-high range and the Reactor Vessel Level Indication System as provided under e. less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE channels for the containment recirculation sump level-narrow range less than the Minimum Channels OPERABLE requirement of Table 3.3-10, restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours. (3)

c. With the number of OPERABLE channels for the main steam line radiation monitor, or the containment area radiation monitor-high range or the plant vent radiation monitor-high range less than the Minimum Channels OPERABLE requirements of Table 3.3-10, initiate the pre-planned alternate method of monitoring the appropriate parameter(s) within 72 hours and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days that provide actions taken, cause of the inoperability and plans and schedule for restoring the channels to OPERABLE status.

g ~~f~~. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-7.

e. With the number of OPERABLE channels for the Reactor Vessel Level Indication System less than the Required Number of Channels or the Minimum Channels OPERABLE requirement of Table 3.3-10 as applicable, restore the inoperable channel(s) to OPERABLE status as specified in respective Action Statement a. or b. if repair is feasible during plant operation. If repair is not feasible, prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2. within 14 days that provides actions taken, cause of the inoperability and plans and schedule for restoring the channel(s) to OPERABLE status. The inoperable channel(s) shall be restored to OPERABLE during the first refueling outage.

f. Action Statement e. applies to first fuel cycle only and statement a. and b. shall become effective thereafter.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure (normal range)	2	1
2. Reactor Coolant Outlet Temperature - T_{hot} (Wide Range)	1/loop in two loops	1/loop in one loop
3. Reactor Coolant Inlet Temperature - T_{cold} (Wide Range)	1/loop in two loops	1/loop in one loop
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	2/steam generator	1/steam generator
8. Refueling Water Storage Tank Water Level	2	1
9. Containment Reactor Cavity (3) Sump Level-Wide Range	2	1
10. Containment Recirculation (1) Sump Level-Narrow Range	N.A.	1
11. Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
12. Reactor Coolant System Subcooling Margin Monitor	1	1
13. PORV Position Indicator	1/valve	1/valve
14. PORV Block Valve Position Indicator	1/valve	1/valve
15. Safety Valve Position Indicator	2*/valve	1/valve
16. In Core Thermocouples	4/core quadrant	2/core quadrant
17. Main Steam Line Radiation Monitor	N.A.	1/steam line
18. Containment Area Radiation Monitor-High Range	N.A.	1
19. Plant Vent Radiation Monitor-High Range	N.A.	1
20. Reactor Vessel Level Indication System	2	1

*One acoustic monitor and one temperature element.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{hot} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{cold} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Refueling Water Storage Tank Water Level	M	R
9. Containment Reactor Cavity Sump Level-Wide Range	M	R
10. Containment Recirculation Sump Level - Narrow Range	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Reactor Coolant System Subcooling Margin Monitor	M	R
13. PORV Position Indicator	M	R
14. PORV Block Valve Position Indicator	M	R
15. Safety Valve Position Indicator	M	R
16. In Core Thermocouples	M	R
17. Main Steam Line Radiation Monitor	M	R
18. Containment Area Radiation Monitor-High Range	M	R*
19. Plant Vent Radiation Monitor-High Range	M	R
20. Reactor Vessel Level Indication System	M	R

*CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

INSTRUMENTATION

CHLORINE DETECTION SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.3.3.7 Two independent Chlorine Detection Systems#, with their Alarm/Trip Setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm, shall be OPERABLE.

APPLICABILITY: All MODES, when bulk chlorine gas is stored on the plant site.

ACTION:

a. With one Chlorine Detection System inoperable, restore the inoperable ~~detection~~ system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room Ventilation System in a recirculation mode with the HEPA filter and charcoal adsorber system in operation.

b. With no Chlorine Detection System OPERABLE, within 1 hour initiate and maintain operation of the Control Room Ventilation System in a recirculation mode with the HEPA filter and charcoal adsorber system in operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each Chlorine Detection System shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days. At least once per 18 months, the following inspections and maintenance shall be performed:

a. Check constant head bottle level and refill as necessary.

b. Clean the sensing cells,

c. Check flow meter operation and clean or replace filters and air lines as necessary,

d. Check air pump for proper operation, and

e. Verify that the detector responds to chlorine.

The Chlorine Detection System is common to both units installed in the normal intakes to the Control Room Ventilation System.

INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.3.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11,

1. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the containment at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

~~2. Restore the inoperable instrument(s) to OPERABLE status within 14 days or in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status, and~~

2

~~2.~~ ~~b.~~ c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. Within 2 hours following a seismic event in excess of 0.02 g, each zone shown in Table 3.3-11 shall be inspected for fires. Within 72 hours an engineering evaluation shall be performed to verify the OPERABILITY of the fire detection system.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.8.2 The supervision circuitry associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

PANEL A

ZONE	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE	
		SMOKE	HEAT OR FLAME
1	Cable Spreading Room	10	$4^{(1)}$
3	4KV Switchgear Bus F Room	1	N.A.
	4KV Switchgear Bus G Room	1	N.A.
	4KV Switchgear Bus H Room	1	N.A.
	4KV Switchgear Ventilation Fan Room	1	N.A.
4	4KV Switchgear Bus F Cable Spreading Area	1	N.A.
	4KV Switchgear Bus G Cable Spreading Area	1	N.A.
	4KV Switchgear Bus H Cable Spreading Area	1	N.A.
7	Containment Electrical Penetration Zone 7	4	N.A.
8	Containment Electrical Penetration Zone 8	3	N.A.
9	Rod Control Programmer/Reactor Trip Breaker Area	3	N.A.
	Battery No. 1 Charger Room	1	N.A.
	Battery No. 2 Charger Room	1	N.A.
	Battery No. 3 Charger Room	1	N.A.
10	480 Volt Bus F Area	1	N.A.
	480 Volt Bus G Area	1	N.A.
	480 Volt Bus H Area	1	N.A.
	Hot Shutdown Panel	1	N.A.
12	Inside Containment	$17^{(2)}$	N.A.
13	Control Room Ventilation Return/Exhaust	1	N.A.
14	Control Room Ventilation Return/Exhaust	1	N.A.
15	Outside the Auxiliary Salt Water Pump Room	1	N.A.

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

PANEL B

ZONE	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE SMOKE	HEAT OR FLAME
1	Residual Heat Removal Pump No. 1 Room	1	N.A.
	Residual Heat Removal Pump No. 2 Room	1	N.A.
2	Component Cooling Water Pump No. 1 Room	1	N.A
	Component Cooling Water Pump No. 2 Room	1	N.A.
	Component Cooling Water Pump No. 3 Room	1	N.A.
	Charging Pump No. 1 Area	1	N.A.
	Charging Pump No. 2 Area	1	N.A.
	Charging Pump No. 3 Area	1	N.A.
	Containment Spray Pump No. 1 Area	1	N.A.
	Containment Spray Pump No. 2 Area	1	N.A.
3	Safety Injection Pump No. 1 Room	1	N.A.
	Safety Injection Pump No. 2 Room	1	N.A.
5	Auxiliary Feedwater Pump No. 1 Area	1	N.A.
	Auxiliary Feedwater Pumps Nos. 2 & 3 Area	1	N.A.
	Boric Acid Transfer Pumps Area	1	N.A.
6	Fire Pumps Area(5)	1	N.A.
	UNIT 2 AUXILIARY Building Supply Fan Room	1	N.A.
	Control Room Ventilation Equipment Room	1	N.A.
7&8	Auxiliary Building Ventilation System Charcoal Filter Bank, EFC-1	12	6
11	Fuel Handling Building Ventilation System Charcoal Filter Bank, EFC-5	6	3
12	Fuel Handling Building Ventilation System Charcoal Filter Bank, EFC-6	6	3
13	Control Room - Control Console	3	N.A.
	Control Room Board	10	N.A.
15	Control Room - Radiation Monitoring	2	N.A.
	Control Room Nuclear Instrumentation	3	N.A.
16(1)	UNIT 1 Auxiliary Building Supply Fan Room	1	N.A.
	Control Room Ventilation Equipment Room	1	N.A.
16(2)	Boric Acid Tanks Area	1(3)	N.A.

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

PANEL B

ZONE	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE SMOKE	MINIMUM INSTRUMENTS OPERABLE HEAT OR FLAME
Not Assigned to Zone	Diesel Generator No. 1 Room	N.A.	2(1)
	Diesel Generator No. 2 Room	N.A.	2(1)
	Diesel Generator No. 3 Room(5)	N.A.	2(1)
Not Assigned to Zone	Solid State Protection System Room	3(4)	N.A.

(1) Heat sensors actuate CO_2 flooding and are tested per Specification 4.7.9.3c. Specifications 4.3.3.8.1 and 4.3.3.8.2 do not apply.

(2) The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

(3) Unit 1 Boric Acid Tank Detectors in Zone 16, Unit 2.

(4) Smoke sensors actuate Halon flooding and are tested per Specification 4.7.9.4c. Specifications 4.3.3.8.1 and 4.3.3.8.2 do not apply.

(5) The Fire Pumps and Diesel Generator No. 3 are common to both units. Located on the Unit 1 side and on the Unit ~~one~~ 1 Fire Detection Instrument Panel only.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION PROCEDURE (ODCP).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3-8.

4.3.3.9.2 At least one salt water pump shall be determined operating ~~at least once per 4 hours to be in operation~~ and providing dilution to the discharge structure whenever dilution is required to meet the ~~site radioactive liquid effluent concentration~~ limits of Specification 3.11.1.1.

(1) status within the time specified in the ACTION, or explain in the next Semi Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.6 why this inoperability was not corrected within the time specified.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. ~~GROSS BETA OR GAMMA~~ RADIOACTIVITY MONITORS PROVIDING Alarm and AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line (RM-18)#	1	40
b. Steam Generator Blowdown Tank (RM-23)	1	41
2. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line (FR-20)#	1	43
b. Steam Generator Blowdown Effluent Lines (FR-53)	1	43
c. Oily Water Separator Effluent Line (FR-251)#	1	43
3. ~~GROSS~~ RADIOACTIVITY MONITOR NOT PROVIDING AUTOMATIC TERMINATION ~~OR~~ of RELEASE		
a. Oily Water Separator Effluent Line (RM-3)#	1	42

This Radioactive Liquid Effluent Monitoring Instrumentation is common to both units.

ACTION STATEMENTS

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue for up to 14 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are analyzed for ~~gross~~ radioactivity (beta or gamma) at a limit of detection of ~~at least~~ 10^{-7} microcuries/~~gram~~ ml:

a. At least once per ~~6~~ 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram DOSE EQUIVALENT I-131.

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per ~~6~~ hours, grab samples are collected and analyzed for ~~gross~~ radioactivity (beta or gamma) at a limit of detection of ~~at least~~ 10^{-7} microcuries/ml or transfer the oily water separator effluent to the Liquid Radwaste Treatment System.

ACTION 43 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves may be used to estimate flow.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. ~~GROSS BETA OR GAMMA~~ (1) RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluents Line (RM-18)	D	P	R(3)	Q(1)
b. Steam Generator Blowdown Tank (RM-23)	D	M	R(3)	Q(1)
2. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line (FR-20)	D(4)	N.A.	R	Q
b. Steam Generator Blowdown Effluent Line (FR-53)	D(4)	N.A.	R	Q
c. Oily Water Separator Effluent Line (FR-251)	D(4)	N.A.	R	Q
3. (1) ~~GROSS~~ RADIOACTIVITY MONITOR NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Oily Water Separator Effluent Line (RM-3)	D	M	R(3)	Q(2)

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint (isolation and alarm), or

b. Relay control circuit failure (isolation only), or

c. Instrument indicates a downscale failure (alarm only), ~~and~~ or

d. Instrument controls not set in operate mode (alarm only).

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure, or

c. Instrument indicates a downscale failure, ~~and~~ or

d. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION~~S~~ FOR OPERATION

3.3.3.10 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.1 and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCP. (1)

APPLICABILITY: As shown in Table 3.3-13

ACTION.

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

b. ~~With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13.~~

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3-9.

(1)

With the number of OPERABLE radioactive gaseous effluent monitoring instrumentation channels less than the Minimum Channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain in the next Semi Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.6 why this inoperability was not corrected within the time specified.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GASEOUS RADWASTE SYSTEM			
Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RM-22)	1	*	50
2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM ~~(for systems not designed to withstand the effects of a hydrogen explosion)~~			
(1) a. Oxygen Monitors (ANR-75 and 76)	2	**	54
~~b. Hydrogen Monitor~~	~~1~~	~~**~~	~~54~~
3. PLANT VENT SYSTEM			
a. Noble Gas Activity Monitor Providing Alarm (RM-14A or 14B)	1	*	52
b. Iodine Sampler	1	*	55
c. Particulate Sampler	1	*	55
d. Flow Rate Monitor (FR-12)	1	*	51
e. Iodine Sampler Flow Rate Monitor	1	*	51
4. STEAM GENERATOR BLOWDOWN TANK VENT SYSTEM			
Gross Activity Monitor (RM-27)	1	*	52
5. CONTAINMENT PURGE SYSTEM			
Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RM-14A ~~and~~ or 14B)	1	(2) ***	53

*** Modes 1-4; also Mode 6 during CORE ALTERATIONS or movement of irradiated fuel within containment.

TABLE 3.3-13 (Continued)

TABLE NOTATIONS

* At all times.

** During GASEOUS RADWASTE SYSTEM operation ~~(treatment for primary system offgases)~~.

(8) ~~*** Whenever Containment Integrity is required during normal operation and refueling.~~

ACTION STATEMENTS

ACTION 50 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tanks contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 51 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 24 hours.

ACTION 52 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per ~~8~~ 12 hours and these samples are analyzed for ~~gross~~ noble activity within ~~8~~ 24 hours.

ACTION 53 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend containment ~~VENTING and~~ PURGING of radioactive effluents via this pathway. Aiir H 3L/3 Or

ACTION 54 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 14 days. With no channels OPERABLE, ~~be in at least HOT STANDBY within 6 hours.~~ operation of this system may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 ...

ACTION 55 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

(1) ~~ACTION 56 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations; otherwise, the hydrogen concentration shall be assumed to exceed 4% by volume and take the ACTION required by Specification 3.11.2.5.~~

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE REQUIRED
1. GASEOUS RADWASTE SYSTEM					
Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RM-22)	P	P	R(3)	Q(1)	*
2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
(1) a Oxygen Monitors (ANR-75 or 76)	D	N.A.	Q(4)	M	**
~~b. Hydrogen Monitor~~	~~D~~	~~N.A.~~	~~Q(4)~~	~~M~~	~~**~~
3. PLANT VENT SYSTEM					
a. Noble Gas Activity Monitor Providing Alarm (RM-14A or 14B)	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W(5)	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W(5)	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor (FR-12)	D	N.A.	R	Q	*
e. Iodine Sampler Flow Rate Monitor	D	N.A.	R	Q	*
4. STEAM GENERATOR BLOWDOWN TANK VENT					
Gross Gamma Activity Monitor (RM-27)	D	M	R(3)	Q(2)	*
5. CONTAINMENT PURGE SYSTEM					
Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RM-14A ~~&~~ 14B) or	D	P	R(1)(3)	Q	***

TABLE 4.3-~~12~~ 9 (Continued)

TABLE NOTATIONS

* At all times.

** During GASEOUS RADWASTE SYSTEM operation ~~(treatment for primary system offgases).~~

*** MODES 1-4, also MODE 6 during CORE ALTERATIONS or movement of irradiated fuel within containment.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the alarm/trip setpoint (isolation and alarm), or

b. Relay control circuit failure (isolation only), or

c. Instrument indicates a downscale failure (alarm only), ~~and~~ or

d. Instrument controls not set in operate mode (alarm only).

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the alarm setpoint, or

b. Circuit failure, or

c. Instrument indicates a downscale failure, ~~and~~ or

d. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. Two volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

(5) The CHANNEL CHECK shall consist of verifying that the iodine cartridge and particulate filter are installed in the sample holders.

~~(6) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:~~

1 ~~a. One to three volume percent hydrogen, balance nitrogen, and~~

~~b. Four volume percent hydrogen, balance nitrogen.~~

INSTRUMENTATION

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITIONS FOR OPERATION

3.3.4.1 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3 (during turbine operation).

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam line inoperable or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore ~~operation may continue for up to 72 hours provided~~ the inoperable valve(s) ~~is restored~~ to OPERABLE status, or ~~at least one valve in the affected steam line is closed; otherwise~~ within 72 hours isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection System otherwise inoperable, within 6 hours ~~either restore the system to OPERABLE status or~~ isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.1.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position:

1) Four high pressure turbine stop valves,

2) Four high pressure turbine control valves,

3) Six low pressure turbine reheat stop valves, and

4) Six low pressure turbine reheat intercept valves.

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position,

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems, and

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION~~S~~ FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.~~*~~

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within ~~1~~ 6 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

~~*See Special Test Exception 3.10.4~~.
~~Specification~~

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION~~S~~ FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant loops in operation when the Reactor Trip System breakers are closed and one reactor coolant loop in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3~~**~~.

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

c. With no reactor 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 reactor coolant loops to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 15% at least once per 12 hours.

(3)

4.4.1.2.3 The required reactor coolant loop(s) shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant pumps may be de-energized for up to 1 hour provided: (1) no operations are permitted which could cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

~~**See Special Test Exception Specification 3.10.4.~~

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.4.1.3 a. At least two of the loops/trains ~~reactor coolant (RC) and/or residual heat removal (RHR) trains~~ listed below shall be OPERABLE (and at least one of these loops/trains shall be in operation)*:

1) a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,**

2) b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,**

3) c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,**

4) d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,**

5) e. Residual Heat Removal (RHR) train 1, and

6) f. Residual Heat Removal (RHR) train 2.

~~b. At least one of the above RC loops or RHR trains shall be in operation.**~~

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required ~~RC~~ loops/train ~~and/or RHR trains~~ OPERABLE, immediately initiate corrective action to return the required loop/train to OPERABLE status as soon as possible; if the remaining OPERABLE loop/train is an RHR train, be in COLD SHUTDOWN within 24 hours.

b. With no ~~RC~~ loop/~~or RHR~~ train 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/train to operation.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 323°F unless: (1) the pressurizer water level is less than 50% or (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

*All reactor coolant pumps and RHR ~~residual 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.

SURVEILLANCE REQUIREMENTS

RHR
and/or ~~residual heat removal~~ pumps

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined ~~to be~~ OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to 15% at least once per 12 hours.
water

4.4.1.3.3 At least one reactor coolant loop or RHR train shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITIONS FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) train shall be OPERABLE and in operation**, and either:

a. One additional RHR train shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 15%.

APPLICABILITY: MODE 5 with Reactor Coolant loops filled##.

ACTION:

a. With one of the RHR ~~loops~~ TRAIN inoperable and ~~less than the above required trains OPERABLE or~~ with less than the required steam generator level, immediately initiate corrective action to return the ~~required~~ inoperable ~~trains~~ train to OPERABLE status or ~~to~~ restore the required steam generator level as soon as possible.

b. With no RHR train 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 RHR train to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 ~~The required RHR train shall be demonstrated~~ OPERABLE ~~pursuant to~~ Specification 4.0.5.

4.4.1.4.1.2 The secondary side water level of at least two steam generators when required, shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.3 At least one RHR train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One RHR train may be inoperable for up to 2 hours for surveillance testing provided the other RHR ~~loop~~ train is OPERABLE and in operation.

3

##A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 323°F unless: (1) the pressurizer water level is less than 50% or (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

**The RHR pump 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.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION~~S~~ FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) trains shall be OPERABLE# and at least one RHR train shall be in operation.*

APPLICABILITY: MODE 5 with ~~R~~eactor ~~C~~oolant loops not filled.

ACTION:

a. With less than the above required RHR trains OPERABLE, immediately initiate corrective action to return the required trains to OPERABLE status as soon as possible.

b. With no RHR train 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 RHR train to operation.

SURVEILLANCE REQUIREMENTS

~~4.4.1.4.2.1 The required RHR trains shall be demonstrated OPERABLE pursuant to Specification 4.0.5.~~

4.4.1.4.2,2 At least one RHR train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One RHR train may be inoperable for up to 2 hours for surveillance testing provided the other RHR train is OPERABLE and in operation.

*The RHR pump 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.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an ~~OPERABLE~~ residual heat removal train into operation. ~~in the shutdown cooling mode~~.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperatures and pressure.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITIONS FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. The provisions of Specification 3.0.4 may be suspended for up to 18 hours per valve for entry into and during operations in MODE 3 for the purpose of setting the pressurizer Code safety valves under ambient (hot) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.3 PRESSURIZER

LIMITING CONDITIONS FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1600 cubic feet and two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring heater group power at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITIONS FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) ~~and remove power from the block valve(s)~~; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the valve through at least one complete cycle of full travel, and

b. Performance of a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel, unless the block valve is closed ~~with power removed~~ in order to meet the requirements of ACTION ~~b~~ a Specification 3.4.4b.

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

LIMITING CONDITIONS FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirement of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

~~Steam Generator Tube Sample Selection and Inspection (Continued)~~

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4.a.8) 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.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

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.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling 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 may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; or

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1) Reactor ~~Primary~~-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

2) A seismic occurrence greater than the Double Design Earthquake, or

3) A loss-of-coolant accident requiring actuation of the engineered safeguards, or

4) A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1) 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;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

7) 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 a Double Design Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

   1) Number and extent of tubes inspected,

   2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

   3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 ~~and require prompt notification of the Commission~~ shall be reported pursuant to Specification 6.9.1 2 prior to resumption of plant operation. ~~The written followup of~~ This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

in a Special Report　　　　within 30 days; and

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % 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.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

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.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug 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 defective tubes and inspect 2S tubes in each other S. G. ~~Prompt~~ notification to NRC pursuant to ~~specification 6.9.1~~ § 50.72(b)(2) of 10CFR Part 50	All other S. G.s are C-1	None	N/A	N/A
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. ~~Prompt~~ notification to NRC pursuant to ~~specification 6.9.1~~ § 50.72(b)(2) of 10CFR Part 50	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Particulate Radioactivity Monitoring System,

b. The Containment Structure Sumps (2) and the Reactor Cavity Sump Level and Flow Monitoring System, and

c. Either the Containment Fan Cooler Collection Monitoring System or the Containment Atmosphere Gaseous Radioactivity Monitoring System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required Leakage Detection Systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required Gaseous and/or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Particulate and Gaseous (if being used) Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment Structure Sumps (2) and the Reactor Cavity Sump Level and Flow Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months, and

c. Containment Fan Cooler Collection Monitoring System (if being used) - performance of CHANNEL FUNCTIONAL TEST at least once per 18 months.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITIONS FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2230 ± 20 psig, and

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual and/or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate or gaseous radioactivity monitor at least once per 12 hours;

b. Monitoring the containment structure sump inventory and discharge at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals at least once per 31 days when the Reactor Coolant System pressure is 2230 ± 20 psig with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours, except when T_{avg} is being changed by greater than 5°F/hour or when diverting reactor coolant to the liquid holdup tank, in which cases the required inventory balance shall be performed within 12 hours after completion of the excepted operation; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. Every refueling outage during startup,

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

c. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve. After each disturbance of the valve, in lieu of measuring leak rate, leak-tight integrity may be verified by absence of pressure buildup in the test line downstream of the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
a. 8948 A, B, C, and D	Accumlator, RHR and SIS first off check valves from RCS cold legs
b. 8819 A, B, C, and D	SIS second off check valves from RCS cold legs
c. 8818 A, B, C, and D	RHR second off check valves from RCS cold legs
d. 8956 A, B, C, and D	Accumulator second off check valves from RCS cold legs
e. 8701* and 8702*	RHR suction isolation valves

*Testing per Specification 4.4.6.2.2c. not required.

REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

LIMITING CONDITIONS FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4:

a. With any one or more chemistry parameter in excess of its Steady-State Limit but within its Transient Limit, restore the parameter to within its Steady-State 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; and

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At all other times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
CHLORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
FLUORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

*Limit not applicable with $T_{avg} \leq 250°F$.

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once per 72 hours
CHLORIDE	At least once per 72 hours
FLUORIDE	At least once per 72 hours

* Not required with $T_{avg} \leq 250°F$

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITIONS FOR OPERATION

3.4.8 The specific activity of the reactor ~~primary~~ coolant shall be limited to:

a. Less than or equal to 1 microCurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/Ē microCuries/gram, of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

The provisions of Specification 3.0.4 are not applicable.

MODES 1, 2 and 3*:

a. With the specific activity of the reactor ~~primary~~ coolant greater than 1 microCurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a ~~primary~~ coolant specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive ~~six~~ month period, prepare and submit a SPECIAL REPORT to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours of operation above this limit. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the reactor ~~primary~~ coolant greater than 1 microCurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

c. With the specific activity of the reactor ~~primary~~ coolant greater than 100/Ē microCuries/gram of gross radioactivity, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5:

a. With the specific activity of the reactor ~~primary~~ coolant greater than 1 microCurie/gram DOSE EQUIVALENT I-131 or greater than 100/Ē microCuries/gram of gross radioactivity, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the ~~primary~~ reactor coolant is restored to within its limits. ~~A REPORTABLE OCCURRENCE~~

*With T_{avg} greater than or equal to 500°F.

ACTION (Continued)

~~shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:~~

1. Reactor power history starting 48 hours prior to the ~~first~~ sample in which the limit was exceeded;

2. ~~Fuel burnup by core region;~~

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded;

4. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and

5. The time duration when the specific activity of the primary coolant exceeded 1 microCurie/gram DOSE EQUIVALENT I-131.

For this ACTION Statement, prepare and Submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, and Chief, Accident Analysis Branch, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555. This Report shall contain the results of the specific activity analysis together with the following information.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the ~~primary~~ reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

2. Results of:

a) the last isotopic analysis for radionuclides performed prior to exceeding the limit,

b) analysis while limit was exceeded, and

c) one analysis after the radioiodine activity was reduced to less than the limit including each isotopic analysis, the date and time of sampling and the radioiodine concentrations;

TABLE 4.4-4

REACTOR
PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Beta Activity Determination**	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for Ē Determination***	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/Ē μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 & 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a ~~one~~ 1 hour period.	1, 2, 3

#Until the specific activity of the Reactor ~~primary~~ Coolant System is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.

** — INSERT

*** — INSERT

**A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available ~~isotopic decay~~ data may be used for pure beta-emitting radio-nuclides.

***A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radio-iodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon these energy peaks identifiable with a 95% confidence level.

Insert

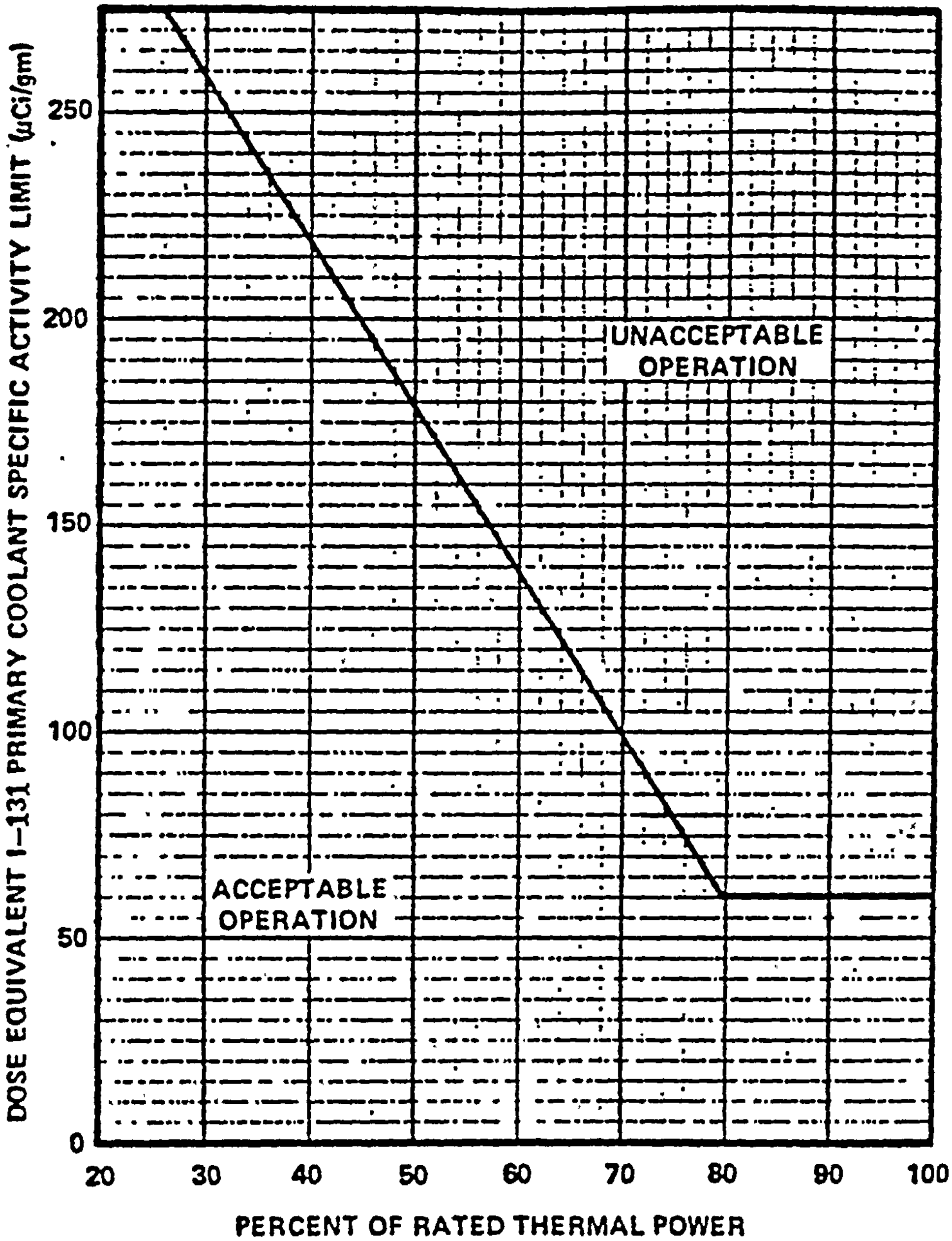

FIGURE 3.4-1

DOSE EQUIVALENT I-131 ~~PRIMARY~~ REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1.0 μCi/GRAM DOSE EQUIVALENT I-131

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any ~~one~~ hour period,

b. A maximum cooldown of 100°F in any ~~one~~ hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits 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 structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per hour during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50, Appendix H, in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3, and the setpoint of Technical Specification 3.4.9.3.a.

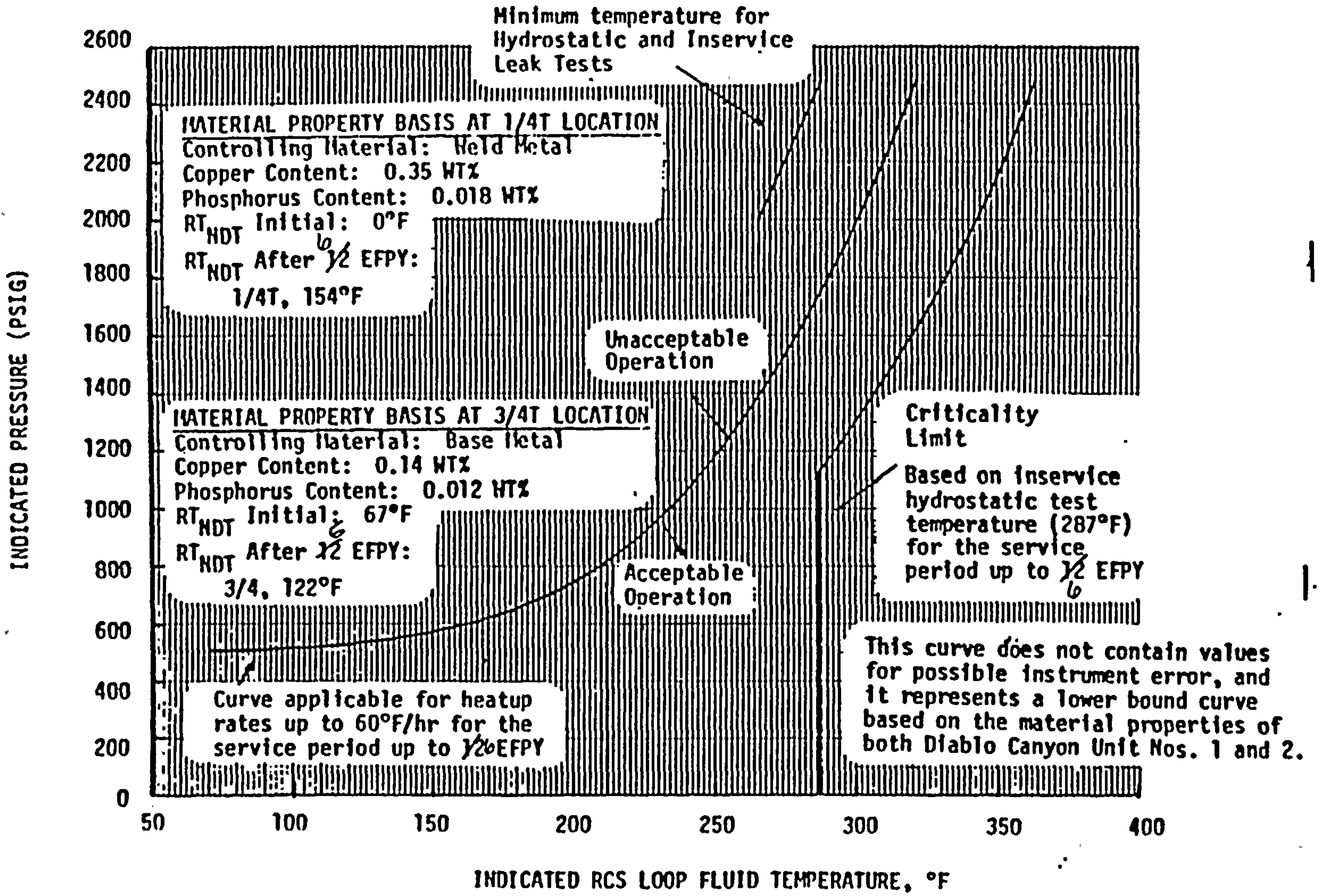

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS

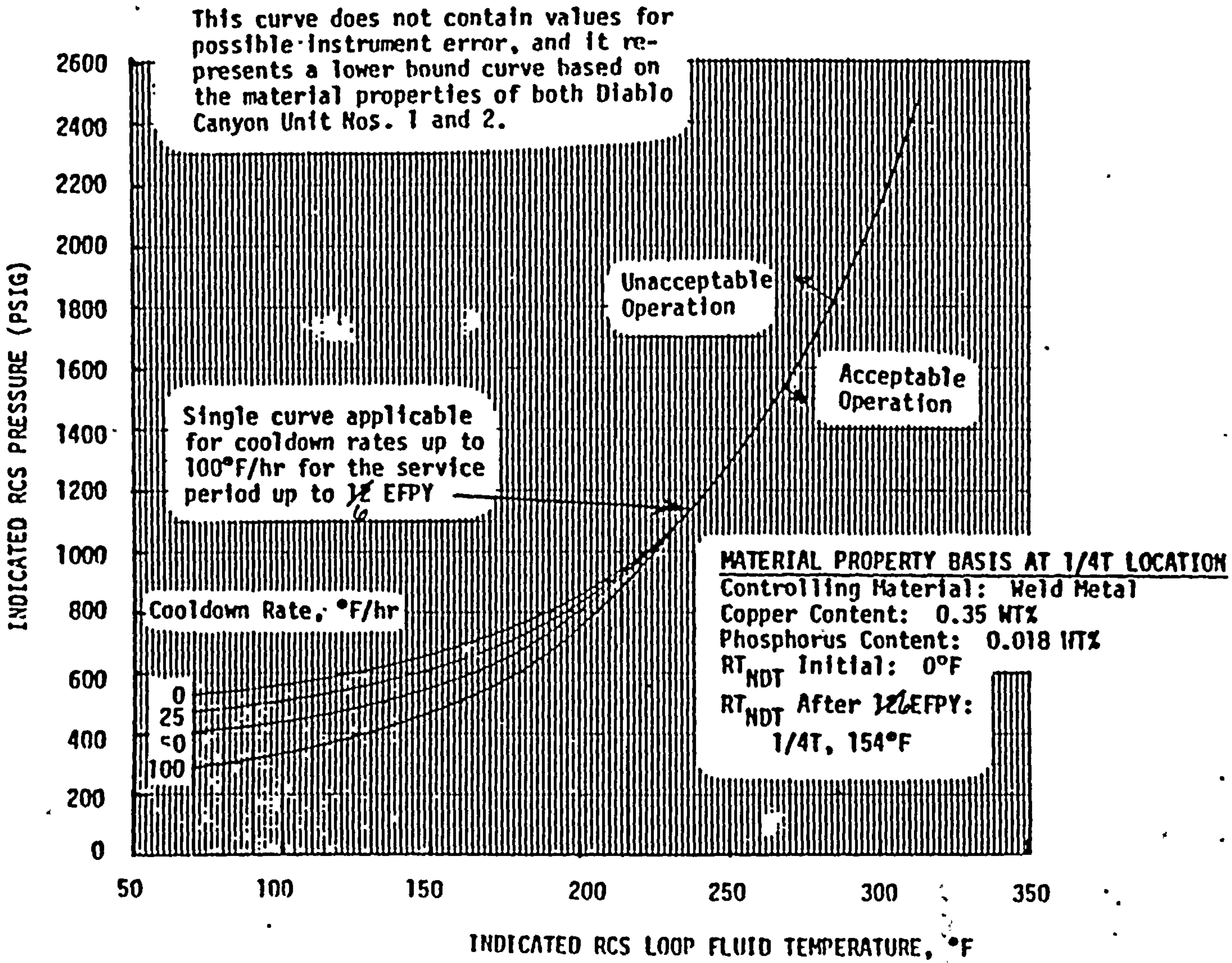

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

UNIT 1

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
S	320°	3.7	First refueling outage
T	140°	3.7	Standby
U	356°	1.1	12
V	184°	1.1	24
W	4°	1.1	38
X	176°	1.1	50
Y	40°	3.7	Standby
Z	220°	3.7	Standby

UNIT 2

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
U	56°	4.8	FIRST REFUELING OUTAGE
X	236°	4.8	3
V	58.5°	4.0	6
Y	238.5°	4.0	10
W	124°	4.8	15
Z	304°	4.8	STANDBY

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION~~S~~ FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any ~~one~~ 1- hour period,

b. A maximum cooldown of 200°F in any ~~one~~ 1- hour period, and

c. A maximum spray water temperature differential of 560°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per hour during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.4.9.3 The following Overpressure Protection Systems shall be OPERABLE:

(1)

~~a. RHR system isolation valves 8701 and 8702 open with power removed from the valve operators when the positive displacement charging pump is in operation, and~~

a ~~b~~. Two power operated relief valves (PORVs) with a lift setting of less than or equal to 450 psig, ~~or~~ insert

~~c.~~ The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.07 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 323°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With the positive displacement charging pump in operation with the RHR isolation valves closed, within one hour either open the RHR isolation valves or secure the positive displacement charging pump.

a. ~~b.~~ With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a 2.07 square inch vent(s) within the next 8 hours.

b. ~~c.~~ With both PORVs inoperable, depressurize and vent the RCS through a 2.07 square inch vent(s) within 8 hours.

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

d. ~~e.~~ The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; ~~and~~.

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

~~d. Testing pursuant to Specification 4.0.5~~.

~~e. Verifying the RHR isolation valves 8701 and 8702 are opened with power removed from the valve operators when the positive displacement charging pump is in operation at least once per 72 hours~~.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

REACTOR COOLANT SYSTEM

3/4.10 STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR VESSEL HEAD VENTS

LIMITING CONDITIONS FOR OPERATION

3.4.11 At least one reactor vessel head vent path consisting of at least two valves in series powered from emergency busses shall be OPERABLE and closed.

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

ACTION:

With the above reactor vessel head vent path inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor vessel head vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

b. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING, and

c. Verifying flow through the reactor vessel head vent paths during venting during COLD SHUTDOWN or REFUELING.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION~~S~~ FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open, and power removed,

b. A contained borated water volume of between 836 and 864 cubic feet of borated water,

a boron concentration of

c. Between 1900 and 2200 ppm ~~of boron~~, and

d. A nitrogen cover-pressure of between 595.5 and 647.5 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within ~~one~~ hour or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within ~~one~~ 6 hours and ~~be~~ in HOT SHUTDOWN within the ~~next 12~~ following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the RCS pressure is above ~~2000~~ 1000 psig by verifying that power to the isolation valve operator is disconnected by sealing the breaker in the open position; and

~~d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically under each of the following conditions:~~

~~1. When an actual or simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) setpoint.~~

~~2. Upon receipt of a safety injection test signal.~~

4.5.1.2 Each accumulator pressure and water level channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of a CHANNEL FUNCTIONAL TEST; and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITIONS FOR OPERATION

3.5.2 Two Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and manually transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once each 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
8703	RHR to RCS Hot Legs	Closed
8802A	Safety Injection To RCS Hot Legs	Closed
8802B	Safety Injection to RCS Hot Legs	Closed
8809A	RHR to RCS Cold Legs	Open
8809B	RHR to RCS Cold Legs	Open
8835	Safety Injection to RCS Cold Legs	Open
8974A	Safety Injection Pump Recir. to RWST	Open
8974B	Safety Injection Pump Recir. to RWST	Open
8976	RWST to Safety Injection Pumps	Open
8980	RWST to RHR Pumps	Open
8982A	Containment Sump to RHR	Closed
8982B	Containment Sump to RHR	Closed
8992	Spray Additive Tank to Eductor	Open
8701	RHR Suction	Closed
8702	RHR Suction	Closed

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) Verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

SURVEILLANCE REQUIREMENTS (Continued)

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by a visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection actuation test signal, and

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal Charging pump,

b) Safety Injection pump, and

c) Residual Heat Removal pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1) Centrifugal Charging pump ≥ 2400 psid,

2) Safety Injection pump ≥ 1455 psid, and

3) Residual Heat Removal pump ≥ 165 psid.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2) At least once per 18 months.

Boron Injection Throttle Valves	Safety Injection Throttle Valves
8810A	8822A
8810B	8822B
8810C	8822C
8810D	8822D

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 346 gpm, and

b) The total pump flow rate is less than or equal to 550 gpm.

2) For safety injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 463 gpm, and

b) The total pump flow rate is less than or equal to 650 gpm.

i. By performing a flow test, during shutdown, following completion of modifications to the RHR system that alter the system flow characteristics, and verifying that with a single pump running, and delivering to all four cold legs, a total flow rate greater than or equal to 3976 gpm.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} Less Than 350°F

LIMITING CONDITIONS FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,*

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 3[illegible]°F.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

Surveillance

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Requirements of Specification 4.5.2.

4.5.3.2 All centrifugal charging pumps and Safety Injection pumps, except the above ~~required~~ allowed OPERABLE pumps, shall be demonstrated inoperable* at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 323°F by verifying that the motor circuit breakers D.C. control power is de-energized.

*An inoperable pump may be made OPERABLE for testing ~~per Specification 4.0.5~~ or for filling accumulators provided the discharge of the pump has been isolated from the Reactor Coolant System by an isolation valve with power removed from the valve operator, or by a sealed closed manual isolation valve.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK

LIMITING CONDITIONS FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A minimum contained borated water volume of 900 gallons of borated water,

A boron concentration of

b. Between 20,000 and 22,500 ppm ~~of boron~~, and

c. A minimum solution temperature of 145°F.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% ~~delta~~ k/k at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume through a recirculation flow test at least once per 7 days,

b. Verifying the boron concentration of the water in the tank at least once per 7 days, and

c. Verifying the water temperature at least once per 24 hours.

EMERGENCY CORE COOLING SYSTEMS

HEAT TRACING

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least two independent channels of heat tracing shall be OPERABLE for the boron injection tank and for the heat traced portions of the associated flow paths.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With only one channel of heat tracing on either the boron injection tank or on the heat traced portion of an associated flow path OPERABLE, operation may continue for up to 30 days provided the tank and flow path temperatures are verified to be greater than or equal to 145°F at least once per 8 hours; otherwise, be in HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.2 Each heat tracing channel for the boron injection tank and associated flow path shall be demonstrated OPERABLE:

a. At least once per 31 days by energizing each heat tracing channel, and

b. At least once per 24 hours by verifying the tank and flow path temperatures to be greater than or equal to 145°F. The tank temperature shall be determined by measurement. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of ~~greater than or equal to~~ 400,000 gallons ~~of borated water~~,

b. A boron concentration of between 2000 and 2200 ppm ~~boron~~, and

c. A minimum solution ~~water~~ temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and
2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside ambient air temperature is less than 35°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 ~~PRIMARY~~ CONTAINMENT

CONTAINMENT INTEGRITY

(3)

LIMITING CONDITIONS FOR OPERATION

3.6.1.1 ~~Primary~~ CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without ~~primary~~ CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within ~~one~~ 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 ~~Primary~~ CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3; ~~and~~

b. By verifying that each containment air lock is in compliance with the requirements ~~OPERABLE per~~ of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at greater than or equal to P_a, 47 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed more often than once per 92 days.

LIMITING CONDITIONS FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to L_a, 0.10 percent by weight of the containment air per 24 hours at P_a, 47 psig, or, *as applicable*,

2) Less than or equal to L_t, 0.0472 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 25.0 psig

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either ~~(a)~~ the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or ~~(b) with~~ the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either greater than or equal to P_a, 47 psig, or at greater than or equal to P_t, 25.0 psig, as applicable, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, as applicable, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, as applicable, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t, as applicable, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1) Confirms the accuracy of the Type A test by verifying that the supplemental test results Lc, minus the sum of the Type A and the superimposed leak, Lo, is equal to or less than ~~difference between supplemental and Type A test data is within~~ 0.25 L_a, or 0.25 L_t, as applicable.

2) Has a duration sufficient to establish accurately the change in leakage between the Type A test and the supplemental test, and

3) Requires ~~the quantity of~~ that the rate at which gas injected into the containment or bled from the containment during the supplemental test ~~to be equivalent to at least 25 percent of the total measured leakage rate at either greater than or equal to P_a, 47 psig, or greater than or equal to P_t, 25.0 psig, or at least four times the measured~~ leakage ~~in any one hour period.~~ is between 0.75 La and 1.25 La, or 0.75 Lt and 1.25 Lt, as applicable,

SURVEILLANCE REQUIREMENTS (Continued)

d. Type B and C tests shall be conducted with gas at P_a, 47 psig, at intervals no greater than 24 months except for tests involving:

   1) Air locks, and

   2) Penetrations using continuous Leakage Monitoring Systems.

e. Air locks shall be tested and demonstrated OPERABLE ~~at~~ by the requirements of Specification 4.6.1.3;

f. Type B tests for penetrations employing a continuous Leakage Monitoring System shall be conducted at greater than or equal to P_a, 47 psig, at intervals no greater than once per 3 years;

g. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated Leakage Measurement System; and

h. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at greater than or equal to P_a, 47 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed,

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next ~~six~~ 6 hours and in COLD SHUTDOWN within the following 30 hours, and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed, restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next ~~six~~ 6 hours and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. After each opening except when the air lock is being used for multiple entries, then at least once each 72 hours, by verifying the seal leakage is less than or equal to 0.01 L_a as determined by precision flow measurement when measured for at least 60 seconds with:

1) The volume between the main air lock seals at a constant pressure greater than or equal to 10 psig, and

2) The volume between the emergency air lock seals at a constant pressure greater than or equal to 10 psig.

b. By conducting overall air lock leakage tests at not less than P_a, 47 psig, and verifying the overall air lock leakage rate is within its limit:

(2)

1. At least once per 6 months*, and

2. Prior to establishing CONTAINMENT INTEGRITY if opened when CONTAINMENT INTEGRITY was not required when maintenance has been performed on the air lock that could affect the air lock sealing capability.**

(1)

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

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*The provisions of Specification 4.0.2 are not applicable.

**This represents an Exemption to Appendix J ~~of 10 CFR 50~~ Paragraph III.D.2 of 10 CFR Part 50.

(3)

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 ~~Primary~~ Containment internal pressure shall be maintained between -~~3.5~~ 1.0 and +0.3 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The ~~primary~~ containment internal pressure shall be determined to be within the limits at least once per 12 hours.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 ~~Primary~~ Containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The ~~primary~~ containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Element and Location

a. TE-85 or TE-86, approximately 100 ft. elevation between crane wall and containment wall,

b. TE-87 or TE-88, approximately 100 ft. elevation between steam generators,

c. TE-89 or TE-90, approximately 140 ft. elevation near equipment hatch or stairs at 270°, respectively, and

d. TE-91 or TE-92, approximately 184 ft. elevation on top of steam generator missile barriers away from steam generators.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.~~1~~2. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

in a Special Report *within 15 days*

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 One purge supply line and/or one purge exhaust line of the Containment Purge System may be open or the vacuum/pressure relief line may be open. The vacuum/pressure relief line may be open provided the vacuum/pressure relief isolation valves are blocked to prevent opening beyond 50° (90° is fully open). Operation with any two of these three lines open is permitted. Operation with the purge supply and/or exhaust isolation valves open or with the vacuum/pressure relief isolation valves open up to 50° shall be limited to less than or equal to 200 hours during a calendar year.

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

ACTION:

With a containment purge supply and/or exhaust isolation valve open or the vacuum/pressure relief isolation valves open up to 50° for more than 200 hours during a calendar year, or the Containment Purge System open and the vacuum/pressure relief lines open, or with the vacuum/pressure relief isolation valves open beyond 50°, close the open isolation valve(s) or isolate the penetration(s) within 1 hour; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 The position of the containment purge supply and exhaust isolation valves and the vacuum/pressure relief isolation valves shall be determined closed at least once per 31 days.

4.6.1.7.2 The cumulative time that the purge supply and/or exhaust isolation valves or the vacuum/pressure relief isolation valves are open during a calendar year shall be determined at least once per 7 days.

4.6.1.7.3 The vacuum/pressure relief isolation valves shall be verified to be blocked to prevent opening beyond 50° at least once per 18 months.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.6.2.1 Two Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring spray function to a RHR system taking suction from the containment sump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By Verifying that on recirculation flow, each pump develops a differential pressure of greater than or equal to 205 psid when tested pursuant to Specification 4.0.5;

c. At least once per 18 months by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a ~~containment isolation~~ Phase 'B' Isolation test signal, and

2) Verifying that each spray pump starts automatically on a ~~containment isolation~~ Phase 'B' Isolation test signal.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank with a contained volume of between 2025 and 4000 gallons of between 30 and 32 ~~percent~~ % by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 6 months by:

1) Verifying the contained solution volume in the tank, and

2) Verifying the concentration of the NaOH solution by chemical analysis.

c. At least once per 18 months by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Spray actuation test signal; and

d. At least once per 5 years by verifying both spray additive and RWST full flow from the test valve 8993 in the Spray Additive System.

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.6.2.3 At least three independent groups of containment fan cooler units shall be OPERABLE with a minimum of two units in two groups and one unit in the third group.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION.

a. With one group of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore the inoperable group of cooling fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required containment cooling fans inoperable, and both Containment Spray Systems OPERABLE, restore at least one group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required containment cooling fans inoperable and one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment fan cooler unit shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Starting each containment fan cooler unit and verifying that each containment fan cooler unit operates for at least 15 minutes,

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying a cooling water flow rate of greater than or equal to 2000 gpm to each cooler, and

3) Verifying that each containment fan cooler unit starts on low speed and the dampers transfer to the accident position.

b. At least once per 18 months by verifying that each containment fan cooler unit starts automatically on a Safety Injection test signal.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and ~~either~~.

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

4.6.3.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that on a Phase "A" ~~containment~~ Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Phase "B" ~~containment~~ Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each Containment Ventilation isolation valve actuates to its isolation position.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.3 The isolation time of each testable power-operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each containment ventilation isolation valve, except the air sample supply and return valves, shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cycling, then at least once per 72 hours, by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Specification 4.6.1.2c. for all other Type B and C penetrations, the combined leakage rate is less than or equal to $0.60L_a$.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
A.1 PHASE "A" ISOLATION VALVES		
FCV-151#	Steam Generator No. 1 Blowdown OC	≤ 10
FCV-154#	Steam Generator No. 2 Blowdown OC	≤ 10
FCV-157#	Steam Generator No. 3 Blowdown OC	≤ 10
FCV-160#	Steam Generator No. 4 Blowdown OC	≤ 10
FCV-244#	Steam Generator No. 4 Sample OC	≤ 10
FCV-246#	Steam Generator No. 3 Sample OC	≤ 10
FCV-248#	Steam Generator No. 2 Sample OC	≤ 10
FCV-250#	Steam Generator No. 1 Sample OC	≤ 10
FCV-253	Reactor Coolant Dr. Tk. PP Disch. Isol. IC	≤ 10
FCV-254	Reactor Coolant Dr. Tk. PP Disch. OC	≤ 10
FCV-255	Reactor Coolant Dr. TK. Vent. Isol. IC	≤ 10
FCV-256	Reactor Coolant Dr. Tk. Vent. Isol. OC	≤ 10
FCV-257	Reactor Coolant Dr. Tk. Sample to GA OC	≤ 10
FCV-258	Reactor Coolant Dr. Tk. Sample to GA IC	≤ 10
FCV-260	Reactor Coolant Dr. Tk. N_2 Supply OC	≤ 10
FCV-361	CCW Return from Excess Letdown HX OC	≤ 10
FCV-500	Containment Sump Discharge Isolation IC	≤ 10
FCV-501	Containment Sump Discharge Isolation OC	≤ 10
FCV-584	Containment Instrument Air Supply OC	≤ 10
FCV-633	Containment Fire Water Isolation OC	≤ 10
FCV-654	Incore Cooler Chilled H_2O Supply OC	≤ 10
FCV-655	Incore Cooler Chilled H_2O Supply IC	≤ 10

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
1. Phase "A" Isolation Valves (Continued)		
FCV-656	Incore Cooler Chilled H_2O Return OC	$\leq$ 10
FCV-657	Incore Cooler Chilled H_2O Return IC	$\leq$ 10
8029	Primary H_2O Supply to Pressurizer Relief Tk. OC	$\leq$ 10
8034A	Pressurizer Relief Tk. to GA IC	$\leq$ 10
8034B	Pressurizer Relief Tk. to GA OC	$\leq$ 10
8045	Pressurizer Relief Tk. N_2 Supply OC	$\leq$ 10
8149A	Letdown Orifice RO-27 Outlet IC	$\leq$ 10
8149B	Letdown Orifice RO-28 Outlet IC	$\leq$ 10
8149C	Letdown Orifice RO-29 Outlet IC	$\leq$ 10
8152	Letdown Line Isolation OC	$\leq$ 10
8871	ECCS Check Valve Test Line IC	$\leq$ 10
8880	Accumulator N_2 Fill OC	$\leq$ 10
8883	ECCS Check Valve Test Line OC	$\leq$ 10
8961	ECCS Check Valve Test Line OC	$\leq$ 10
9354A	Pressurizer Steam Space Sample IC	$\leq$ 10
9354B	Pressurizer Steam Space Sample OC	$\leq$ 10
9355A	Pressurizer Liquid Space Sample IC	$\leq$ 10
9355B	Pressurizer Liquid Space Sample OC	$\leq$ 10
9356A	RCS Hot Leg Sample IC	$\leq$ 10
9356B	RCS Hot Leg Sample OC	$\leq$ 10
9357A	Accumulator Sample IC	$\leq$ 10
9357B	Accumulator Sample OC	$\leq$ 10
8100	RCP Seal Water Return OC	$\leq$ 10
8112	RCP Seal Water Return IC	$\leq$ 10

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
B. PHASE "B" ISOLATION VALVES		
FCV-356	CCW Supply to RCP's and Support Coolers OC	~~NA~~ N.A.
FCV-357	RCP Thermal Barrier CCW Return OC	~~NA~~ N.A.
FCV-363	RCP Oil Cooler/Support Cooler CCW Return OC	~~NA~~ N.A.
FCV-749	RCP Oil Cooler/Support Cooler CCW Return IC	~~NA~~ N.A.
FCV-750	RCP Thermal Barrier CCW Return IC	~~NA~~ N.A.
C. CONTAINMENT VENTILATION ISOLATION VALVES		
FCV-660##	Containment Purge Supply IC	≤ 2
FCV-661##	Containment Purge Supply OC	≤ 2
FCV-662	Containment Vacuum/Pressure Relief IC	≤ 10
FCV-663	Containment Pressure Relief OC	≤ 10
FCV-664	Containment Vacuum Relief OC	≤ 10
FCV-678	Containment Air Sample Supply IC	≤ 10
FCV-679	Containment Air Sample Supply OC	≤ 10
FCV-681	Containment Air Sample Return OC	≤ 10
RCV-11##	Containment Purge Exhaust IC	≤ 2
RCV-12##	Containment Purge Exhaust OC	≤ 2.
D. MANUAL VALVES		
AIR-I-1-585*	Instrument Air Supply to Containment (FCV-584 Bypass) OC	~~NA~~ N.A.
AIR-S-1-200*	Service Air Supply to Containment OC	~~NA~~ N.A.

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
4. Manual Valves (continued)		
AXS-X-26*	Aux. Steam Supply to Containment OC	~~NA~~ N.A.
CS-X-31~~0~~	Containment Spray to Misc. Equipment Drain Tank OC	~~NA~~ N.A.
CS-X-32~~0~~	Containment Spray to Misc. Equipment Drain Tank OC	~~NA~~ N.A.
FW-X-140#	Auxiliary Feedwater to Stm. Gen. No. 1 OC	~~NA~~ N.A.
FW-X-147#	Auxiliary Feedwater to Stm. Gen. No. 2 OC	~~NA~~ N.A.
FW-X-153#	Auxiliary Feedwater to Stm. Gen. No. 3 OC	~~NA~~ N.A.
FW-X-157#	Auxiliary Feedwater to Stm. Gen. No. 4 OC	~~NA~~ N.A.
MS-X-902#	Nitrogen to Steam Generators OC	~~NA~~ N.A.
RCS-X-512~~0~~*	Miscellaneous Equipment Drain Tank Isolation Valve OC	~~NA~~ N.A.
SI-X-161*	Isolating Valve F~~I~~ I-927 OC	~~NA~~ N.A.
VAC-X-1*	Containment Hydrogen Purge Supply Fan No. 1 and External H_2 Recombiner to Containment OC	~~NA~~ N.A.
VAC-X-2*	Containment Hydrogen Purge Supply Fan No. 2 and External H_2 Recombiner to Containment OC	~~NA~~ N.A.
8767	Refueling Cavity to Refueling Water Purification Pump OC	~~NA~~ N.A.
8787	Refueling Water Purification Pump to Refueling Cavity OC	~~NA~~ N.A.
8795	Refueling Cavity to Refueling Water Purification Pump IC	~~NA~~ N.A.
8796	Refueling Water Purification Pump to Refueling Cavity IC	~~NA~~ N.A.
8969#	Charging Pump to S.I. Test Line OC	~~NA~~ N.A.
PEN-X-65A*#	Main Airlock Equalizing Valve to Atmosphere	~~NA~~ N.A.
PEN-X-65B*#	Main Airlock Equalizing Valve to Containment	~~NA~~ N.A.

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
Manual Valves (Continued)		
PEN-1-66A*#	Emergency Airlock Equalizing Valve to Atmosphere	~~NA~~ N.A.
PEN-1-66B*#	Emergency Airlock Equalizing Valve to Containment	~~NA~~ N.A.
F. POWER OPERATED VALVES		
FCV-22#	No. 4 Stm. Gen. Mn. Steam Isol. Valve Bypass OC	~~NA~~ N.A.
FCV-23#	No. 3 Stm. Gen. Mn. Steam Isol. Valve Bypass OC	~~NA~~ N.A.
FCV-24#	No. 2 Stm. Gen. Mn. Steam Isol. Valve Bypass OC	~~NA~~ N.A.
FCV-25#	No. 1 Stm. Gen. Mn. Steam Isol. Valve Bypass OC	~~NA~~ N.A.
FCV-37#	Auxiliary FWP Turb. Steam Supply S/G No. 2 OC	~~NA~~ N.A.
FCV-38#	Auxiliary FWP Turb. ~~Stm.~~ Steam Supply S/G No. 3 OC	~~NA~~ N.A.
FCV-41#	No. 1 Stm. Generator Mn. Steam Isol. OC	≤ 5
FCV-42#	No. 2 Stm. Generator Mn. Steam Isol. OC	< 5
FCV-43#	No. 3 Stm. Generator Mn. Steam Isol. OC	< 5
FCV-44#	No. 4 Stm. Generator Mn. Steam Isol. OC	≤ 5
FCV-235*	Containment H_2 Sample Supply IC	~~NA~~ N.A.
FCV-236*	Containment H_2 Sample Supply OC	~~NA~~ N.A.
FCV-237*	Containment H_2 Sample Return OC	~~NA~~ N.A.
FCV-238*	Containment H_2 Sample Supply IC	~~NA~~ N.A.
FCV-239*	Containment H_2 Sample Supply OC	~~NA~~ N.A.
FCV-240*	Containment H_2 Sample Return OC	~~NA~~ N.A.
FCV-658	Containment Purge to Aux. Bldg. Filters/ Ext. H_2 Recombiners Supply IC	~~NA~~ N.A.
FCV-668	Containment Purge to Aux. Bldg. Filters/Ext. H_2 Recombiner Supply OC	~~NA~~ N.A.

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
Power Operated Valves (Continued)		
FCV-659	Containment Purge to Purge System Filters/Ext. H_2 Recombiners Supply IC	~~NA~~ N.A.
FCV-669	Containment Purge to Purge System Filters/Ext. H_2 Recombiners Supply OC	~~NA~~ N.A.
FCV-760#	Steam Generator No. 1 Blowdown IC	~~NA~~ N.A.
FCV-761#	Steam Generator No. 2 Blowdown IC	~~NA~~ N.A.
FCV-762#	Steam Generator No. 3 Blowdown IC	~~NA~~ N.A.
FCV-763#	Steam Generator No. 4 Blowdown IC	~~NA~~ N.A.
FCV-696*	Reactor Cavity Sump Sample (Post LOCA) Supply IC	~~NA~~ N.A.
FCV-697*	Reactor Cavity Sump Sample (Post LOCA) Supply OC	~~NA~~ N.A.
FCV-698*	Containment Air Sample (Post LOCA) Supply IC	~~NA~~ N.A.
FCV-699*	Containment Air Sample (Post LOCA) Supply OC	~~NA~~ N.A.
FCV-700*	Containment Air Sample (Post LOCA) Return OC	~~NA~~ N.A.
PCV-19#	Steam Generator No. 1 10% Atmosphere Steam Dump OC	~~NA~~ N.A.
PCV-20#	Steam Generator No. 2 10% Atmosphere Steam Dump OC	~~NA~~ N.A.
PCV-21#	Steam Generator No. 3 10% Atmosphere Steam Dump OC	~~NA~~ N.A.
PCV-22#	Steam Generator No. 4 10% Atmosphere Steam Dump OC	~~NA~~ N.A.
8107#	Charging Line Isolation OC	~~NA~~ N.A.
8700A#	RCS Hot Leg to RHR Pump 1 OC	~~NA~~ N.A.
8700B#	RCS Hot Leg to RHR Pump 2 OC	~~NA~~ N.A.
8701#	RCS Loop 4 Hot Leg to RHR IC	~~NA~~ N.A.
8703#	RHR to RCS Hot Legs 1 and 2 IC	~~NA~~ N.A.
8716A#	RHR to RCS Hot Legs OC	~~NA~~ N.A.
8716B#	RHR to RCS Hot Legs OC	~~NA~~ N.A.
8801A#	Boron Injection Tank Discharge to RCS OC	~~NA~~ N.A.

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
5. Power Operated Valves (Continued)		
8801B#	Boron Injection Tank Discharge to RCS OC	NA N.A.
8802A#	Safety Injection to RCS Hot Legs OC	NA N.A.
8802B#	Safety Injection to RCS Hot Legs OC	NA N.A.
8809A#	Residual Heat Removal to RCS Cold Legs 1 and 2	NA N.A.
8809B#	Residual Heat Removal to RCS Cold Legs 3 and 4	NA N.A.
8823#	Safety Injection Check Valve Test Line IC	NA N.A.
8824#	Safety Injection Check Valve Test Line IC	NA N.A.
8843#	Boron Injection Tank to Cold Leg Check Valve Test Line IC	NA N.A.
8835#	Safety Injection to RCS Cold Legs OC	NA N.A.
8885A#	RHR to Cold Leg Test Line IC	NA N.A.
8885B#	RHR to Cold Leg Test Line IC	NA N.A.
8982A#	Containment Sump to Residual Heat Removal Train 1 OC	NA N.A.
8982B#	Containment Sump to Residual Heat Removal Train 2 OC	NA N.A.
8980#	Refueling Water Storage Tank to RHR OC	NA N.A.
9001A	Containment Spray Pump No. 1 Isolation OC	NA N.A.
9001B	Containment Spray Pump No. 2 Isolation OC	NA N.A.
9003A#	Residual Heat Removal to Containment Spray OC	NA N.A.
9003B#	Residual Heat Removal to Containment Spray OC	NA N.A.
6. CHECK VALVES		
8028	Relief Valve Outlets to Pressurizer Relief Tank IC	NA N.A.
8046	Primary Water to Pressurizer Relief Tank IC	NA N.A.
8047	Nitrogen to Pressurizer Relief Tank IC	NA N.A.
8109	Seal Water Return IC	NA N.A.

TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
8368A thru 8368D	Seal Water to Reactor Coolant Pumps IC	~~NA~~ N.A.
8916	Nitrogen Supply to Accumulators IC	~~NA~~ N.A.
9011A	Containment Spray IC	~~NA~~ N.A.
9011B	Containment Spray IC	~~NA~~ N.A.
CCW-585 ~~Check Valve~~	CCW Supply to RCP IC	~~NA~~ N.A.
CCW-591 ~~Check Valve~~	CCW Return from RCP (FCV-749 Bypass) IC	~~NA~~ N.A.
CCW-670 ~~Check Valve~~	CCW Return from RCP (FCV-750 Bypass) IC	~~NA~~ N.A.
MS-5208 ~~Check Valve~~#	Nitrogen Supply to Stm. Gen. IC*	~~NA~~ N.A.
CCW-695 ~~Check Valve~~	CCW Supply to Excess Letdown Heat Exchanger OC	~~NA~~ N.A.
VAC-200 ~~Check Valve~~	Containment Hydrogen Purge Supply IC	~~NA~~ N.A.
VAC-201 ~~Check Valve~~	Containment Hydrogen Purge Supply IC	~~NA~~ N.A.
VAC-116 ~~Check Valve~~	Containment Air Sample (Post LOCA) Return IC	~~NA~~ N.A.
LWS-60 ~~Check Valve~~	Nitrogen Supply to Reactor Coolant Drain Tank IC	~~NA~~ N.A.
AIR-I-587 ~~Check Valve~~	Instrument Air Supply IC	~~NA~~ N.A.
AIR-S-114 ~~Check Valve~~	Service Air Supply IC	~~NA~~ N.A.
VAC-21 ~~Check Valve~~	Containment Air Sample Return IC	~~NA~~ N.A.
AXS-208 ~~Check Valve~~	Auxiliary Stm. Supply to Containment IC	~~NA~~ N.A.
FP-180 ~~Check Valve~~	Containment Fire Water IC - Unit 1 only	~~NA~~ N.A.
FP-867	Containment Fire Water IC - Unit 2 only	
VAC-252 ~~Check Valve~~	Containment H_2 Sample Return IC	~~NA~~ N.A.
VAC-253 ~~Check Valve~~	Containment H_2 Sample Return IC	~~NA~~ N.A.

*May be opened on an intermittent basis under administrative control (Normally closed manual or remotely OPERABLE valves only)
#Not subject to Type C leakage tests
##The provisions of Specification 3.0.4 are not applicable.

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS/MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers/monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen analyzer/monitor inoperable, restore the inoperable analyzer/monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen analyzers/monitors inoperable, restore at least one analyzer/monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analyzer/monitor shall be demonstrated OPERABLE at least once per 92 days by performing a CHANNEL CALIBRATION using a zero and span gas.

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITIONS FOR OPERATION

3.6.4.2 Two independent ~~containment~~ Hydrogen Recombiner Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Hydrogen Recombiner System inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Hydrogen Recombiner System shall be demonstrated OPERABLE:

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; and

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 all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more main steam line Code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in ~~COLD~~ HOT SHUTDOWN within the following ~~30~~ 6 hours.

b. In MODE 3 a maximum of 19 safety valves may be made inoperable to permit insitu testing of the ~~operable~~ safety valve as required by Specification 4.7.1.1.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (Percent of RATED THERMAL POWER)
1	87 *
2	64 *
3	42 *

* Unless the Reactor trip System breakers are in the open position.

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

LIFT SETTING (± 1%)*	ORIFICE SIZE
1065 psig	4.515 inches
1078 psig	4.515 inches
1090 psig	4.515 inches
1103 psig	4.515 inches
1115 psig	4.515 inches

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.7.1.2 At least three steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate vital busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per ~~92~~ 31 ~~31~~ days by:

1) Verifying that each motor-driven pump develops a differential pressure of greater than or equal to 1370 psid on recirculation flow;

SURVEILLANCE REQUIREMENTS (Continued)

2) Verify that the steam turbine-driven pump develops a differential pressure of greater than or equal to 1312 psid on recirculation flow when the secondary steam supply pressure is greater than 650 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3; and

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months by verifying that each auxiliary feedwater pump starts and valve opens* as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal.

*For the steam turbine-driven pump, when the secondary steam supply pressure is greater than 650 psig.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained volume of at least 178,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the CST inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the fire water tank common to Units 1 and 2 as a backup supply to the auxiliary feedwater pumps and restore the CST to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The CST shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The fire water tank shall be demonstrated OPERABLE at least once per 12 hours by verifying that there is at least 270,000 gallons in the tank and that the fire water tank to auxiliary feedwater pump flow path is OPERABLE whenever the fire water tank is the supply source for the auxiliary feedwater pumps.

~~*See Special Test Exceptions Specification 3.10.4.~~

PLANT SYSTEMS

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.10 microcurie/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.10 microcurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radio~~Activity~~ Determination*	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) ~~1~~ Once per 31 days, whenever the gross radioactivity determination indicates ~~iodine~~ concentrations greater than 10% of the allowable limit for radioiodines. b) ~~1~~ Once per 6 months, whenever the gross radioactivity determination indicates ~~iodine~~ concentrations less than or equal to 10% of the allowable limit for radioiodines.

* A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the secondary coolant except for radionuclides with half-lives less than 15 minutes. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITIONS FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODES 1: With one ~~main steam line isolation valve~~ MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise ~~reduce power to less than or equal to 5 percent of RATED THERMAL POWER within~~ be in HOT STANDBY 6 hours, within the next [illegible] and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3: With one ~~main steam line isolation valve~~ MSIV inoperable, subsequent operation in MODES 2 or 3 may proceed provided the isolation valve is maintained closed.

a. The isolation valve is maintained closed.

b. ~~The provisions of Specification 3.0.4 are not applicable.~~

Otherwise, be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each ~~main steam line isolation valve~~ MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the reactor ~~primary~~ and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the ~~primary~~ reactor or secondary coolant is less than 70°F.

PLANT SYSTEMS

3/4.7.3 VITAL COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two vital component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one vital component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two vital component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position; and

b. At least once per 18 months by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection test signal or ~~containment isolation~~ phase "B" [illegible] test signal, as appropriate.

PLANT SYSTEMS

3/4.7.4 AUXILIARY SALTWATER SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.7.4.1 At least two auxiliary saltwater trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one auxiliary saltwater train OPERABLE, restore at least two trains to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1. At least two auxiliary saltwater trains shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

3/4.7.5 CONTROL ROOM VENTILATION SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.7.5.1 The Control Room Ventilation System* shall be OPERABLE** with two separate trains with each train consisting of one main supply fan, one filter booster fan and one pressurization supply fan, and one HEPA Filter and Charcoal Adsorber System.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

With one Control Room Ventilation System train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one Control Room Ventilation System train inoperable, restore the inoperable train to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE Control Room Ventilation System train in the recirculation mode.

b. With both Control Room Ventilation System trains inoperable, or with the OPERABLE Control Room Ventilation System required to be in the recirculation mode by ACTION a. not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.5.1 Each Control Room Ventilation System train shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F;

b. At least once per 31 days by:

1) Initiating flow through the HEPA Filter and Charcoal Adsorber System and verifying that each booster fan and pressurization supply fan operates for at least 10 continuous hours with the heaters operating,

2) Verifying that each Ventilation System redundant fan is aligned to receive electrical power from a separate OPERABLE vital bus, and

3) Starting (unless already operating) each main supply fan and verifying that it operates for 15 minutes.

** *The system may be considered OPERABLE with no chlorine monitors provided no bulk chlorine gas is stored ~~on the plant site~~ within the SITE BOUNDARY.

* The Control Room Ventilation System is common to both units

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in 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 2100 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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%, and

3) Verifying a system flow rate of 2100 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

d. After 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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3.5 inches Water Gauge while operating the system at a flow rate of 2100 cfm ± 10%;

2) Verifying that on a Phase "A" Isolation test signal, the system automatically switches into the pressurization mode of operation with approximately 27% (determined by damper position) of the flow through the HEPA filters and charcoal adsorber banks;

3) Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during the pressurization mode of system operation; and

SURVEILLANCE REQUIREMENTS (Continued)

4) Verifying that the heaters dissipate 5 ± 1 kW when tested in accordance with ANSI N510-1975. 8c (1c)

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 2100 cfm ± 10%; and 8c (1c)

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon test gas while operating the system at a flow rate of 2100 cfm ± 10%. 8c

PLANT SYSTEMS

3.4.7.6 AUXILIARY BUILDING SAFEGUARDS AIR FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two Auxiliary Building Safeguards Air Filtration System exhaust trains with one common HEPA filter and charcoal adsorber bank and at least two exhaust fans shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the HEPA filter and charcoal adsorber bank inoperable, restore the HEPA filter and charcoal adsorber bank to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With only one exhaust fan OPERABLE, restore at least two exhaust fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Each Auxiliary Building Safeguards Air Filtration System train shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Initiating flow through the HEPA filter and charcoal adsorber bank and verifying that the train operates for at least 10 continuous hours with the heaters operating, and

2) Verifying that each exhaust fan is aligned to receive electrical power from a separate OPERABLE vital bus.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by:

1) Verifying no detectable leakage through the Auxiliary Building Safeguards Air Filtration System Dampers M2A and M2B when subjected to a bubble test at a pressure of greater than or equal to 30 inches Water Gauge;

2) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in 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 73,500 cfm ± 10%.

3) 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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 6%; and

4) Verifying a system flow rate of 73,500 cfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975~~ 80.

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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 6%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3.7 inches water Gauge while operating the system at a flow rate of 73,500 cfm ± 10%.

2) Verifying that flow is established through the HEPA filter and charcoal adsorber bank on a Safety Injection test signal, and

3) Verifying that the heaters dissipate 50 ± 5 kW when tested in accordance with ANSI N510-~~1975~~ 80.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-~~1975~~ 80 for a DOP test aerosol while operating the system at a flow rate of 73,500 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-~~1975~~ 80 for a halogenated hydrocarbon test gas while operating the system at a flow rate of 73,500 cfm ± 10%.

PLANT SYSTEMS

3/4.7.7 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.7.1 All snubbers shall be OPERABLE. 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.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.7.1g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in lieu of the requirements of Specification 4.0.5.

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 are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that type shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given type shall be performed in accordance with the following schedule:

SURVEILLANCE REQUIREMENTS (Continued)

No. of Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify: (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.7.1f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

c. Transient Event Inspection

A visual inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data. This inspection shall be performed within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

*The inspection interval of each type of snubber shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found.

#The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers of each type shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected for each snubber type prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.7.1f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.7.1f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

SURVEILLANCE REQUIREMENTS (Continued)

e. Functional Tests (Continued)

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location 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 test 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.

f. 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;

2) Snubber bleed, or release rate where required, is present in both tensions and compression, within the specified range;

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions 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.

g. Functional Test Failure Analysis

An engineering 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.

SURVEILLANCE REQUIREMENTS (Continued)

g. Functional Test Failure Analysis (Continued)

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 design service.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.7.1e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

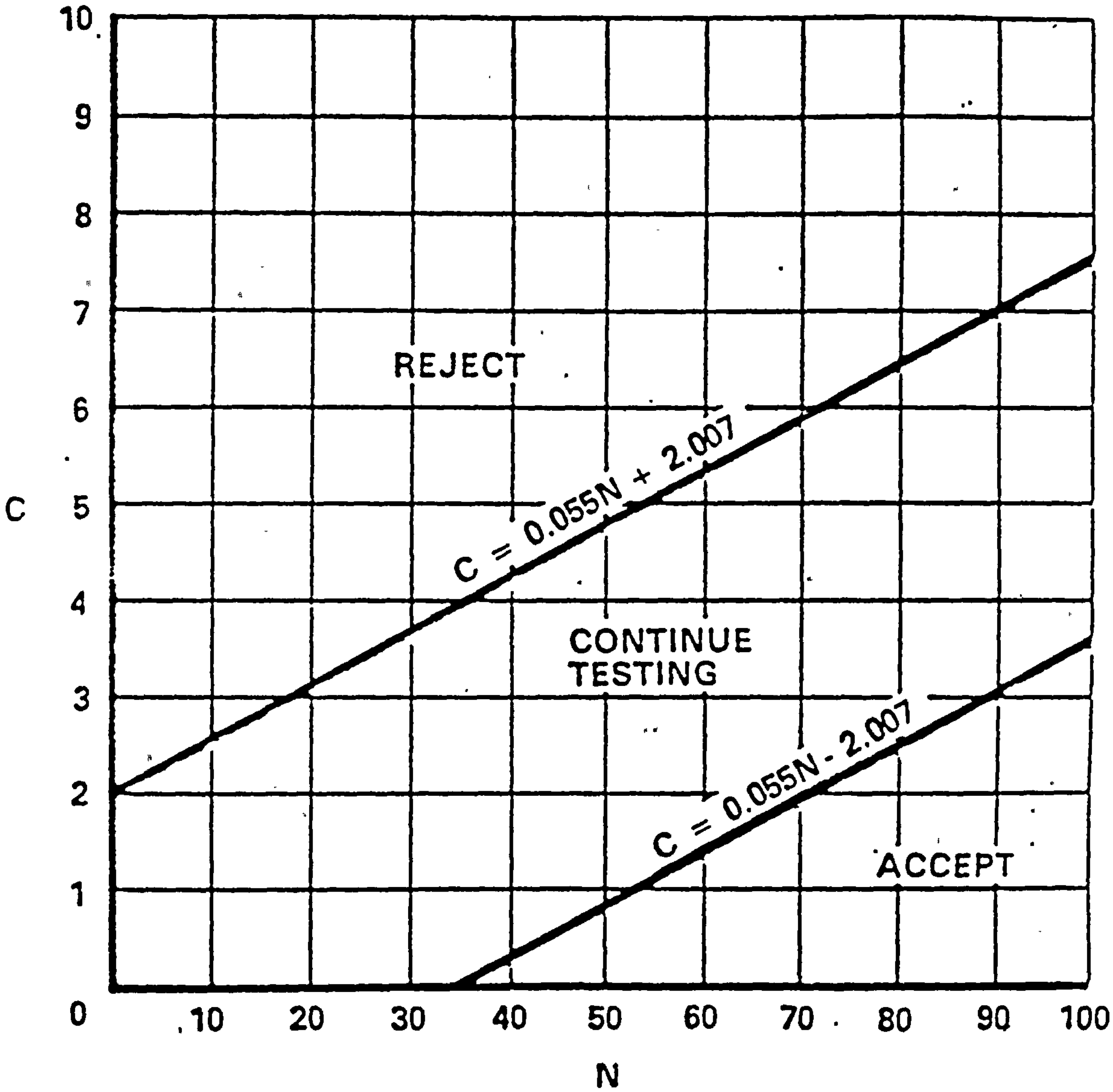

FIGURE 4.7-1

SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITIONS FOR OPERATION

3.7.8.1 Each sealed source containing radioactive material in excess of 100 microCuries of beta and/or gamma emitting material or 10 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use, and either:

1. Decontaminate and repair the sealed source, or

2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.1.1 Test Requirements - Each sealed source shall be tested for leakage and or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.8.1.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

1) With a half-life greater than 30 days (excluding Hydrogen 3), and

2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in core components or the core and following repair or maintenance of the source.

4.7.8.1.3 Reports - A report shall be prepared and submitted to the Commission on an annual bases if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

3/4.7.9 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.7.9.1 The Fire Suppression Water System shall be OPERABLE with;

a. At least two high pressure pumps, each with a capacity of 2250 gpm, taking suction from the firewater tank containing a minimum usable volume of 270,000 gallons, with its discharge aligned to the fire suppression* header,

b. Gravity feed to the fire suppression header from a raw water reservoir with a minimum usable volume of 270,000 gallons, and

c. An OPERABLE fire suppression header flow path through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler, hose standpipe or Spray System riser required to be OPERABLE per Specifications 3.7.9.2 and 3.7.9.5.

APPLICABILITY: At all times.

ACTION:

a. With either one pump and its water supply or the raw water gravity feed water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or, ~~in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to restore the inoperable equipment to OPERABLE status or to~~ provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire Suppression Water System otherwise inoperable,

1. Establish a backup Fire Suppression Water System within 24 hours, ~~and~~

~~2. In lieu of any other report required by Specification 6.9.1, submit a Special Report in accordance with Specification 6.9.2:~~

~~a) By telephone within 24 hours,~~

~~b) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and~~

~~c) In writing 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.~~

*Except the suppression header check valve bypass valves shall be closed unless the gravity feed to the suppression header is inoperable, or inadequate to supply the system for fire protection purposes.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.9.1 The Fire Suppression Water System shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the water supply volume,

b. At least once per 31 days by starting each pump and operating it for at least 15 minutes on recirculation flow,

c. At least once per 31 days by verifying that each valve* (manual, power-operated or automatic) in the flow path is in its correct position. (1)

d. At least once per 6 months by performance of a system flush of the ~~except that the fire suppression water system header serving fire hose stations in the containment as listed in Table 3.7.4 of Specification 3.7.6.5 need only be flushed when in MODE 5 or 6 for more than 24 hours~~. outside distribution loop to verify no flow package;

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

f. At least once per 18 months by:

1) Verifying that each pump develops at least 2250 gpm at a developed head of 80 psig.

2) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3) Verifying that each high pressure pump starts to maintain the fire Suppression Water System pressure greater than or equal to 67 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

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(1) * Except valves which are located inside the containment and are locked, sealed, or otherwise secured in position. These valves shall be verified in the correct position during each COLD SHUTDOWN except such verification need not be performed more often than once per 92 days.

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.7.9.2 The following Spray and/or Sprinkler Systems shall be OPERABLE: (1)

a. 500/25 kV Main Transformer Bank,

b. 25/4 kV Unit Auxiliary Transformer No. 2,

c. 230/12 kV Standby Startup Transformer No. 1,

d. 12/4 kV Standby Startup Transformer No. 2,

e. Auxiliary Feedwater Pumps ~~[illegible]~~ Area, ~~and~~

g. ~~f.~~ Component Cooling Water Pumps Area, and

f. ~~g.~~ Centrifugal Charging Pump Area, ~~and~~

h. Containment Penetration Area.

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour, establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged, for other areas, establish an hourly fire watch patrol. ~~Restore the system to OPERABLE status within 14 days or in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.~~

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path is in its correct position,

b. At least once per 12 months by cycling each testable valve in the flow path through at least one cycle,

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months:

1) By performing a system functional test which includes simulated automatic actuation of the system, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one cycle.

2) By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air or water flow test through each open head spray/sprinkler/deluge header and verifying each open head spray/sprinkler/deluge nozzle is unobstructed.

CO2 SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9.3 The Low Pressure CO_2 System with the subsystems shown in Table 3.7-3 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the Low Pressure CO_2 System is required to be OPERABLE.

ACTION:

a. With the above required automatic Low Pressure CO_2 System inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. With a hose reel station inoperable, provide backup fire suppression equipment in the affected area within 1 hour. ~~Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.~~

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.3 The above required Low Pressure CO_2 System shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying CO_2 storage tank level to be $\geq$ 50% and pressure to be $\geq$ 290 psig,

b. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path is in its correct position, and

c. At least once per 18 months by verifying:

   1) The system valves and associated ventilation dampers and fire door release mechanisms actuate manually and automatically, upon receipt of a simulated actuation signal, and

   2) Flow from each nozzle during a "Puff Test."

TABLE 3.7-3

CO_2 SYSTEM

1\. ~~a.~~ Diesel Generator No. 1 flooding subsystem.

2\. ~~b.~~ Diesel Generator No. 2 flooding subsystem.

(3) 3. ~~c.~~ Diesel Generator No. 3 flooding subsystem. (common to both Units)

4\. ~~d.~~ Cable spreading room flooding subsystem.

5\. ~~e.~~ CO_2 hose reel subsystem stations:

~~1) No. 1 or No. 2 in 4 & 12 kV switchgear room.~~
~~2) No. 3 or No. 4 serving vital 4 kV cable spreading rooms.~~
~~3) No. 5 or No. 6 serving vital 4 kV switchgear rooms,~~
~~4) No. 8 serving vital 480 V load centers F, G, H,~~
~~5) No. 10 serving battery rooms, and~~
~~6) No. 12 serving battery rooms.~~

LOCATION	UNIT 1	UNIT 2
4 & 12 kV Switchgear Room	No. 1 or 2	No. 13 or 14
Vital 4 kV Cable Spreading Rooms	No. 3 or 4	No. 15 or 16
Vital 4 kV Switchgear Rooms	No. 5 or 6	No. 17 or 18
Vital 480 V Load Centers F, G and H	No. 8	No. 9
Battery Rooms	No. 10	No. 12
Rod Control Programmer Room	No. 11	No. 11

PLANT SYSTEMS

HALON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9.4 The Protection and Safeguards Actuation System Room Halon System shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the Halon System is required to be OPERABLE.

ACTION:

a. With the above required Halon System inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment. ~~for those areas in which redundant systems or components could be damaged. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.~~

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.4 The above required Halon System shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying Halon storage tank weight to be at least 95% of full charge weight and pressure to be be at least 90% of full charge pressure,

b. At least once per 18 months by:

1) Verifying the system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a simulated test signal, and

2) Performance of a flow test through headers and nozzles to assure no blockage.

c. Prior to the expiration of its service life and at least once per 5 years by replacement of the explosive initiator with an explosive initiator from a certified manufacturing lot.

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITIONS FOR OPERATION

3.7.9.5 The fire hose stations shown in Table 3.7-4 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire water hose stations shown in Table 3.7-4 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise, route the additional hose within 24 hours. ~~Restore the fire hose station to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and plans and schedule for restoring the station to OPERABLE status~~.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.5 Each of the fire hose stations shown in Table 3.7-4 shall be demonstrated OPERABLE:

a. At least once per 31 days, by visual inspection of the stations accessible during plant operations to assure all required equipment is at the station,

b. At least once per 18 months, by:

   1) Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

   2) Removing the hose for inspection and re-racking, and

   3) Inspecting all gaskets and replacing any degraded gaskets in couplings.

c. At least once per 3 years by:

   1) Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

   2) Conducting a hose hydrostatic test at a pressure of 300 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

TABLE 3.7-4

FIRE HOSE STATIONS

LOCATION	ELEVATION	IDENTIFICATION UNIT 1	UNIT 2
1. Yard			
Near Fence North of Main Bank	95	YL-5	N.A.
Near Fence North of No. 1 Startup Bank	85	YL-6	YL-8
Near Fence North of No. 2 Startup Bank for Unit 1 South of No. 2 Startup Bank for Unit 2	85	YL-7	YL-19
Near Main Transformer	85	YL-9	YL-20
Between Main Transformer & Aux. Boiler	85	YL-10	N.A.
South of Diesel Generator Rooms	85	N.A.	YL-18
2. Auxiliary Building			
Near West End of Hall for Unit 1 Near East End of Hall for Unit 2	64	FW-A1-1	FW-A6-2
Outside RHR Pump Room	64	FW-A2-1	FW-A5-2
Near Containment Spray Pumps	73	FW-A8-1	FW-A10-2
Near No. 3 CCW Pump	73	FW-A7-1	FW-A9-2
Outside Letdown HX Room	85	FW-A14-1	FW-A16-2
Penetration Area Near Doorway to Secondary Sample Room	85	FW-A11-1	FW-A17-2
Fuel Handling Bldg. North of Door to Auxiliary Feedpumps for Unit 1 South of Door to Auxiliary Feedpumps for Unit 2	100	FW-A21-1	FW-A30-2
Penetration Area	100	FW-A22-1	FW-A28-2
Penetration Area	100	FW-A23-1	FW-A29-2
Near Door to VCT Room	100	FW-A24-1	FW-A25-2
North of Boric Acid Transfer Pumps for Unit 1 South of Boric Acid Transfer Pumps for Unit 2	100	FW-A25-1	FW-A27-2
Penetration Area	115	FW-A35-1	FW-A42-2
Penetration Area	115	FW-A36-1	FW-A41-2
Stairwell East of Cable Spreading Room (Common to both Units)	127	FW-A55-1	FW-A55-1

TABLE 3.7-4 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	IDENTIFICATION UNIT 1	UNIT
3. Containment			
Near Main Airlock	140	FW-C7-1	FW-C7-.
Near Equipment Hatch	140	FW-C6-1	FW-C6-:
Near Emergency Airlock	140	FW-C5-1	FW-C5-.
Near CFCU No. 1	140	FW-C8-1	FW-C8-:
Near ECCS Sump	91	FW-C3-1	FW-C3-:
135°, outer wall	91	FW-C2-1	FW-C2-2
4. Turbine Building			
Between Diesels and 12/4 KV room	85	FW-T49-1	FW-T52-2
Corridor Outside Diesel Mufflers for Unit 1 Near SW Evaporator for Unit 2	104	FW-T50-1	FW-T23-2
Near Unit 1 Passenger Elevator, Near Unit 2 Freight Elevator	104	FW-T19-1	FW-T20-2
North of Switchgear Vent Fans for Unit 1 South of Switchgear Vent Fans for Unit 2	119	FW-T51-1	FW-T54-2
5. Intake Structure			
Wall outside ASW Pump 1 Room	(-) 2	FW-3-~~I2~~ I2-1	FW-3-I3-1
6. Fuel Handling Building			
Outside, near east entrance to Aux. Building Filters	115	FW-A53-1	FW-A58-2
Outside, near entrance to Fuel Handling Building Filters for Unit 1. Inside Fuel Handling Building for Unit 2	140	FW-A52-1	FW-A51-2

PLANT SYSTEMS

3/4.7.10 FIRE BARRIER PENETRATIONS

LIMITING CONDITIONS FOR OPERATION

3.7.10 All fire barrier penetrations (including cable penetration barriers, firedoors and fire dampers) in fire ~~zone~~ area boundaries protecting safety related areas shall be functional.

APPLICABILITY: ~~At all times.~~ Whenever the equipment protected by the fire barrier penetrations is required to be OPERABLE.

ACTION:

a. With one or more of the above required fire barrier penetrations non-functional, within 1 hour either, establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol. ~~Restore the non-functional fire barrier penetration(s) to functional status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the non-functional penetration and plans and schedule for restoring the fire barrier penetration(s) to functional status.~~

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10 Each of the above required fire barrier penetrations shall be verified to be functional:

a. At least once per 18 months by a visual inspection.

b. Prior to returning a fire barrier penetrations to functional status following repairs or maintenance by the performance of a visual inspection of the affected fire barrier penetrations.

(3)

PLANT SYSTEMS

3/4.7.11 AREA TEMPERATURE MONITORING

LIMITING CONDITION~~S~~ FOR OPERATION

3.7.11 The temperature of each area shown in Table 3.7-5 shall ~~be maintained~~ not be exceeded within the ~~limits indicated in Table 3.7-5~~. for more than 8 hours or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-5/

~~a.~~ For more than 8 hours, prepare and submit ~~a Special Report~~ to the Commission ~~pursuant to Specification 6.9.2~~ within ~~the next~~ 30 days, pursuant to Specification 6.9.2, a Special Report that provid~~ing~~es a record of the amount by which and the cumulative time the temperature in the affected area(s) exceeded ~~its~~ the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

b. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-5 By more than 30°F, ~~in addition to the~~ prepare and submit a Special Report as required by ACTION a above, and within 4 hours either restore the area(s) to within ~~its~~ the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.11 The temperature in each of the areas shown in Table 3.7-5 shall be determined to be within its limit at least once per 12 hours.

TABLE 3.7-5

AREA TEMPERATURE MONITORING

	AREA	TEMPERATURE LIMIT (°F)
1.	4 kV Vital Bus F Area	≤ 103
2.	4 kV Vital Bus G Area	≤ 103
3.	4 kV Vital Bus H Area	≤ 103
4.	480 V Bus F Area	≤ 103
5.	480 V Bus G Area	≤ 103
6.	480 V Bus H Area	≤ 103
7.	Battery No. 1 Room	≤ 89
8.	Battery No. 2 Room	≤ 89
9.	Battery No. 3 Room	≤ 89
10.	Instrument Inverter No. 1 Room	≤ 103
11.	Instrument Inverter No. 2 Room	≤ 103
12.	Instrument Inverters No. 3 & 4 Room	≤ 103
13.	Diesel Generator No. 1 Room	≤ 129
14.	Diesel Generator No. 2 Room	≤ 129
15.	Diesel Generator No. 3 Room (Common to both Units)	≤ 129
16.	Cable Spreading Room	≤ 103
17.	Safety Injection Pump No. 1 Room	≤ 103
18.	Safety Injection Pump No. 2 Room	≤ 103
19.	Centrifugal Charging Pump No. 1 Room	≤ 103
20.	Centrifugal Charging Pump No. 2 Room	≤ 103
21.	Reciprocating Charging Pump No. 3 Room	≤ 103
22.	Residual Heat Removal Pump No. 1 Room	≤ 103
23.	Residual Heat Removal Pump No. 2 Room	≤ 103
24.	Containment Spray Pumps Nos. 1 & 2 Area	≤ 103
25.	Motor Auxiliary Feedwater Pump No. 2 Room	≤ 103
26.	Motor Auxiliary Feedwater Pump No. 3 Room	≤ 103
27.	Auxiliary Feedwater Valves LCV-113 & 115 Area	≤ 103
28.	Component Cooling Water Pump No. 1 Room	≤ 111
29.	Component Cooling Water Pump No. 2 Room	≤ 111
30.	Component Cooling Water Pump No. 3 Room	≤ 111

3/4.7.12 ULTIMATE HEAT SINK

LIMITING CONDITIONS FOR OPERATION

3.7.12 The ultimate heat sink (UHS)* shall be OPERABLE with an inlet water temperature of less than or equal to 64°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the requirements of the above specification not satisfied, place a second vital component cooling water heat exchanger in service within 8 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12 The UHS shall be determined OPERABLE by verifying the inlet water temperture to be within its limit:

a. At least once per 24 hours when the inlet water tempe^a^rture is equal to or less than 60°F, or

b. At least once per 12 hours when the inlet water temperature is greater than 60°F but less than 62°F, or

c. At least once per 2 hours when the inlet water temperature is equal to or greater than 62°F but less than or equal to 64°F.

---

* The UHS is common to both Units.

PLANT SYSTEMS

3/4.7.13 FLOOD PROTECTION

LIMITING CONDITIONS FOR OPERATION

3.7.13 The full length of both breakwaters (east and west)* shall be above the mean lower low water (MLLW) level.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any of the breakwaters length reduced to less than MLLW level, prepare and submit to the Commission within 90 days pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of how the degradation occurred and if the breakwaters are continuing to degrade,

   2. A planned course of action to repair the damage and the schedule for accomplishing the repair, and

   3. Summary description of action(s) to be taken to prevent a recurrence.

   The provisions of Specification 3.0.4 are not applicable.

b. With 500 feet or more of the breakwater length reduced to less than MLLW level, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.13.1 The breakwaters shall be visually inspected at least once per 31 days during the months of November through April and the full length of each shall be verified above MLLW level.

4.7.13.2 The crest of the breakwaters shall be visually inspected for settlement and displacement at least once per 12 months. Photographs shall be taken during the inspection, from a sufficient vantage point, to show the approximate condition of both breakwaters.

4.7.13.3 The results of any visual inspection as well as surveys that detect damage and the associated photograph(s) shall be included in the May Monthly Operating Report in accordance with Specification 6.9.1.10.

* Both breakwaters are common to both Units

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

Two supply trains of the Diesel Fuel Oil Storage and Transfer System with a combined storage of 31,023 gallons of fuel for one unit operation and 52,046 gallons of fuel for two unit operation.

LIMITING ~~CONDITIONS~~ CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two independent circuits (one with delayed access) between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. Three separate and independent diesel generators*, each with:

1) A separate engine-mounted fuel tank containing a minimum volume of 200 gallons of fuel, and

2) ~~A common fuel storage system containing a minimum volume of 31,023 gallons of fuel, and two diesel fuel transfer pumps.*~~ ~~for one Unit operation and 52,046 gallons of fuel for two unit operation~~

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing ~~Surveillance~~ the required of Specification ~~Requirements~~ 4.8.1.1.1a. and 4.8.1.1.2a.2) within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and three diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing ~~Surveillance~~ the required of Specification ~~Requirements~~ 4.8.1.1.1a. and 4.8.1.1.2a.2) within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and three diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~*One inoperable fuel transfer pump is equivalent to one inoperable diesel generator.~~

OPERABILITY of the third (common) diesel generator shall include the capability of functioning as a power source for the required Unit & upon automatic demand from that Unit &.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITIONS FOR OPERATION

ACTION (Continued)

c. With one diesel generator inoperable in addition to ACTION a. or b. above verify that:

   1. All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generators as a source of emergency power are also OPERABLE, and

   2. When in MODE 1, 2, or 3 that at least two auxiliary feed pumps are OPERABLE.

   If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performing the requirements of Specification 4.8.1.1.2a.2) within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two or more of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing the requirements of Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least two of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least three diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With one supply train of the Diesel Fuel Oil Storage and Transfer System inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and be in HOT SHUTDOWN within the following 6 hours.

g. With both supply trains of the Diesel Fuel Oil Storage and Transfer System inoperable, restore at least one supply train, including the common storage system, to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and be COLD SHUTDOWN within the following 30 hours.

Diesel Generator 3 is common to both units and
be tested more frequently than required to satisfy
OPERABILITY requirement for the most limiting uni
Unit Specific portions of this Surveillance Requi
Shall be performed on an alternating schedule
Units 1 and 2. Testing of Diesel Generator 3 on o
unit does not place the other unit in an ACT
St.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by:

1) Transferring 4 KV vital bus power supply from the normal circuit to the alternate circuit (manually and automatically) and to the delayed access circuit (manually), and

2) Verifying that on a Safety Injection test signal, without loss of offsite power, the preferred, immediate access offsite power source energizes the emergency busses with permanently connected loads and energizes the auto-connected emergency (accident) loads through sequencing timers.

4.8.1.1.2 Each diesel generator* shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:**

1) Verifying the fuel level in the engine-mounted fuel tank;

2) Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 13 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals; ~~with startup on each signal verified at least once per 92 days~~:

a) Manual, or

b) Simulated loss of offsite power by itself (Startup bus under voltage), or

c) A Safety Injection actuation test signal by itself.

3) Verifying the generator is synchronized, loaded to greater than or equal to 2484 kw in less than or equal to 60 seconds, and operates for greater than or equal to 60 minutes;

4) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses;*** and

5) Verifying the diesel engine protective relay trip cutout switch is returned to the cutout position following each diesel generator test.

[See Insert]

Insert p. 3/4 8-3

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*Tests of Diesel Generator 3 to satisfy the frequency specified in Table 4.8-1 and in Surveillance Requirement 4.8.1.1.2.b for one Unit ~~1~~ may be counted in determining whether the frequency specified in Table 4.8-1 and in Surveillance Requirement 4.8.1.1.2b for the other Unit ~~2~~ is satisfied. Unit-specific portions of this Surveillance Requirement for Diesel Generator 3 shall be performed on an alternating schedule with signals from Units 1 and 2.

**All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Further, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures (e.g., gradual acceleration and/or gradual loading > 150 sec) as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

*** May be the associated bus in the other Unit ~~X~~ if that Unit ~~X~~ is in Mode 1, 2, 3 or 4.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months, during shutdown, by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service;

2) Verifying that the load sequence timers are OPERABLE with each load sequence timer within the limits specified in Table 4.8-2;

3) Verifying the generator capability to reject a load of greater than or equal to 508 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 3 Hz;

4) Verifying the generator capability to reject a load of greater than or equal to 2484 kW without tripping. The generator voltage shall not exceed 4580 volts during and following the load rejection;

5) Simulating a loss of offsite power by itself, and:

a) Verifying de-energization of the emergency busses and load shedding from the emergency busses, and

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the required auto-connected loads through sequencing timers and operates for greater than or equal to 5 minutes while its generator is loaded with the permanent and auto-connected loads. After energization of these loads, the steady state voltage and frequency of the emergency buses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

6) Verifying that on a Safety Injection test signal without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 13 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test;

SURVEILLANCE REQUIREMENTS (Continued)

7) Simulating a loss of offsite power in conjunction with a Safety Injection test signal, and

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses;

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through sequencing timers and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization of these loads, the steady state voltage and frequency of the emergency buses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test; and

   c) Verifying that all automatic diesel generator trips, except engine overspeed, low lube oil pressure and generator differential, are bypassed when the diesel engine trip cutout switch is in the cutout position and the diesel is aligned for automatic operation.

8) Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 2750 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 2484 kW. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 13 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24 hour test, perform Specification 4.8.1.1.2b.5)b);*

9) Verifying that the auto-connected loads to each diesel generator do not exceed the maximum rating of 2750 kW;

10) Verifying the diesel generator's capability to:

   a) Synchronize its isolated bus with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

*~~The requirement to verify the 10-second startup and loading of the diesel generator may be waived for the first inspection interval.~~ (14)
If specification 4.8.1.1.2b.5)b) is not satisfactorily completed, it is not necessary to repeat the preceeding 24-hour test. Instead the diesel generator may be operated at 2484 kw for 1 hour or until operating temperature has stabilized.

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~~b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks;~~ by

b ~~c~~. At least once per 31 ~~92~~ days by checking for and removing accumulated water from the fuel oil storage tanks;

c ~~d~~. By sampling new fuel oil in accordance with ASTM-D4057 prior to addition to storage tanks and:

1) By verifying in accordance with the tests specified in ASTM-D975-81 prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

←insert→

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification;

c) A flash point equal to or greater than 125°F; and

d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.

2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.

d ~~e~~. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A.

SURVEILLANCE REQUIREMENTS (Continued)

11) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal opens the auxiliary transformer breaker and automatically sequences the emergency loads onto the diesel generator; and

12) Verifying that the Shutdown Relay lockout feature prevents diesel generator starting only when required:

a) Generator differential current-high, or
b) Engine lube oil pressure-low, or
c) Emergency stop button actuated, or
d) Overspeed trip actuated.

c. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all diesel generators simultaneously, during shutdown, and verifying that all diesel generators accelerate to at least 900 rpm in less than or equal to 10 seconds.

d. At least once per 31 days and after each operation of the diesel ~~when~~ where the period of operation was greater than or equal to 1 hr by checking for and removing accumulated water from the day tank.

4.8.1.1.3 The Diesel Fuel Oil Storage and Transfer System shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Verifying the fuel level in the fuel storage tank, and

2) Verifying that each fuel transfer pump starts and transfers fuel from the storage system to each engine-mounted tank via installed lines.

~~b. At least once per 92 days by verifying that a continuous sample obtained in accordance with ASTM-D270-1975, while the storage tank is on recirculation, has a water and sediment content of less than or equal to 0.05 volume percent and a kinematic viscosity at 40°C of greater than or equal to 1.9 but less than or equal to 4.1 when tested as specified in ASTM-D975-77,~~

~~c. By verifying that a sample obtained in accordance with ASTM-D270-1975 has a:~~

~~1) Water and sediment content of less than 0.05 volume percent and a kinematic viscosity at 40°C of greater than or equal to 1.9 but less than or equal to 4.1 when tested as specified in ASTM-D975-77 prior to addition of new fuel to the storage tanks; and~~

~~2) Impurity level of less than 2 mg. of insolubles per 100 ml. when tested in accordance with ASTM-D2274-70 within 14 days after addition of new fuel to the storage tanks.~~

SURVEILLANCE REQUIREMENTS (Continued)

(1) e/ ~~g~~. At least once per 10 years by:

1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and

2) Performing a visual examination of accessable piping during an operating pressure leak test.

4.8.1.1.4 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.~~1~~2 Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures (on a per nuclear unit basis) in the last 100 valid tests is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b (of Regulatory Guide 1.108, Revision 1, August 1977).

(2) as a Special Report within 30 days

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 100 Valid Tests*	Test Frequency
≤ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
≥ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this ~~test~~ schedule, only valid tests conducted ~~of the OL issuance date~~ (~~shall be included in the computation of the~~) "last 100 valid tests." ~~Entry into this test schedule shall be made at the 31 day test frequency.~~

after the completion of the preoperational test requirements of Regulatory Guide 1.108, Revision 1, August 1977.

STET

TABLE 4.8-2a

LOAD SEQUENCING TIMERS
ESF TIMERS

	COMPONENT	TIMER SETTINGS (~~SECS~~ s) MINIMUM	NOMINAL	MAXIMUM
1. BUS F	Centrifugal Charging Pump No. 1	1.5	2	3
	Safety Injection Pump No. 1	5	6	7
	Containment Fan Cooler Unit No. 2	9	10	11
	Containment Fan Cooler Unit No. 1	13	14	15
	Component Cooling Water Pump No. 1	17	18	19.5
	Auxiliary Saltwater Pump No. 1	20.8	22	23.2
	Auxiliary Feedwater Pump No. 3	24.5	26	28
2. BUS G	Centrifugal Charging Pump No. 2	1.5	2	3
	Residual Heat Removal Pump No. 1	5	6	7.5
	Containment Fan Cooler Unit No. 3	9	10	11.5
	Containment Fan Cooler Unit No. 5	13	14	15
	Component Cooling Water Pump No. 2	17	18	19
	Auxiliary Saltwater Pump No. 2	20.8	22	23.2
	Containment Spray Pump No. 1	24.5	26	29
3. BUS H	Safety Injection Pump No. 2	1	2	3
	Residual Heat Removal Pump No. 2	5	6	7
	Containment Fan Cooler Unit No. 4	9	10	11
	Component Cooling Water Pump No. 3	12.5	14	15
	Auxiliary Feedwater Pump No. 2	17	18	19.5
	Containment Spray Pump No. 2	20.8	22	23.2

TABLE 4.8-2 6 ~~(Continued)~~

LOAD SEQUENCING TIMERS
AUTO TRANSFER TIMERS

	COMPONENT	TIMER SETTINGS (S ~~SEC~~)		
		MINIMUM	NOMINAL	MAXIMUM
1. BUS F	Component Cooling Water Pump No. 1	4	5	6
	Auxiliary Saltwater Pump No. 1	9	10	11
	Auxiliary Feedwater Pump No. 3	13	14	15
	Centrifugal Charging Pump No. 1	18.5	20	21.5
	Containment Fan Cooler Unit No. 1	23.5	25	27
	Containment Fan Cooler Unit No. 2	23.5	25	27
2. BUS G	Component Cooling Water Pump No. 2	4	5	6
	Auxiliary Saltwater Pump No. 2	9	10	11
	Centrifugal Charging Pump No. 2	18.5	20	21.5
	Containment Fan Cooler Unit No. 3	23.5	25	27
	Containment Fan Cooler Unit No. 5	23.5	25	27
3. BUS H	Component Cooling Water Pump No. 3	4	5	6
	Auxiliary Feedwater Pump No. 2	13	14	15
	Containment Fan Cooler Unit No. 4	22	25	27

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One diesel generator with:

1) An engine-mounted fuel tank containing a minimum volume of 200 gallons of fuel,

2) ~~A fuel storage system containing a minimum volume of 8000 gallons of fuel, and~~ One supply train of the Diesel Fuel Oil Storage and Transfer System with storage of 8000 gallons of fuel in addition to the fuel required for the other Unit.

~~3) A fuel transfer pump.~~

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel or ~~crain~~ crane operations with loads over the fuel storage pool. In addition, when in MODE 5 with the Reactor Coolant Loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2, 4.8.1.1.3 and 4.8.1.1.4, except for Specifications 4.8.1.1.1b.2) and 4.8.1.1.2a.2)c), b.2) for ESF timers, b.6), b.~~8~~ 7), b.10), and b.~~12~~ 11).

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITIONS FOR OPERATION

3.8.2.1 The following electrical busses shall be energized in the specified manner:

a. 4160 volt Vital Bus F, b ~~d~~. 480 volt Vital Bus 1F,

c ~~b~~. 4160 volt Vital Bus G, d ~~e~~. 480 volt Vital Bus 1G,

e ~~c~~. 4160 volt Vital Bus H, f. 480 volt Vital Bus 1H,

g. 120 volt Vital Instrument A.C. Bus 11 energized from its associated inverter connected to D.C. Bus 11,*

h. 120 volt Supplemental Vital Instrument A.C. Bus 11A energized from its associated inverter connected to D.C. Bus 11,*

i. 120 volt Vital Instrument A.C. Bus 12 energized from its associated inverter connected to D.C. Bus 12,*

j. 120 volt Vital Instrument A.C. Bus 13 energized from its associated inverter connected to D.C. Bus 13,*

k. 120 volt Supplemental Vital Instrument A.C. Bus 13A energized from its associated inverter connected to D.C. Bus 13,*

l. 120 volt Vital Instrument A.C. Bus 14 energized from its associated inverter connected to D.C. Bus 12,*

m. 125 volt D.C. Bus 11 energized from Battery Bank 11, and its associated ~~battery~~ full capacity charger

n. 125 volt D.C. Bus 12 energized from Battery Bank 12, and its associated ~~battery~~ full capacity charger,

o. 125 volt D.C. Bus 13 energized from Battery Bank 13, and its associated ~~battery~~ full capacity charger

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

ACTION:

a. With one of the required 4160 volt and/or associated 480 volt vital busses not energized, re-energize them within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one Vital Instrument A.C. Bus not energized from its associated inverter, or with one inverter not connected to its associated D.C. Bus, re-energize the Vital Instrument A.C. Bus from an alternate source within 2 hours or be in at least HOT STANDBY within the next 6 hours

*Two Vital Instrument A.C. inverters or one Vital and one Supplemental Vital Instrument A.C. inverter may be disconnected from their D.C. Busses for up to 24 hours for the purpose of performing an equalizing charge on their associated battery bank provided: (1) their vital busses are energized, and (2) the vital busses associated with the other battery banks are energized from their associated inverters and connected to their associated D.C. Buses.

LIMITING CONDITIONS FOR OPERATION

ACTION (Continued)

and in COLD SHUTDOWN within the following 30 hours; re-energize the Vital Instrument A.C. Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. Bus not energized from its associated Battery Bank and full-capa charger, re-energize it from its associated Battery Bank and full-capacity charger within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With one Supplemental Vital Instrument A.C. Bus not energized from its associated inverter or with its inverter not connected to its associated D.C. Bus, re-energize the Supplemental Vital Instrument A.C. Bus from an alternate source within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; re-energize the Supplemental Vital Instrument A.C. Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.8.2.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One 4160 volt and its associated 480 volt A.C. Vital Bus,

b. Two 120 volt Vital Instrument A.C. Busses and one 120 volt Supplemental Vital Instrument A.C. Bus energized from their associated inverters connected to their respective D.C. Busses, and

c. One 125 volt D.C. Bus energized from its associated battery bank, and full-capacity charger.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

3/4.8.3 D.C. SOURCES

OPERATING

LIMITING CONDITIONS FOR OPERATION

3.8.3.1 The following D.C. electrical sources shall be OPERABLE:

a. 125-volt D.C. Battery No. 1 and an associated full-capacity charger,

b. 125-volt D.C. Battery No. 2 and an associated full-capacity charger, and

c. 125-volt D.C. Battery No. 3 and an associated full-capacity charger.

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

ACTION:

With one of the required battery banks and/or full-capacity chargers inoperable, restore the inoperable battery bank and/or full-capacity charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-3 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 130-volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with D.C. bus voltage below 118-volts, or battery overcharge with D.C. bus voltage above 145-volts, by:

1) Verifying that the parameters in Table 4.8-3 meet the Category B limits,

2) Verifying there is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than ~~250~~ 150 x 10^{-6} ~~ohms~~ ohm*, and

3) Verifying that the average electrolyte temperature of 10 of the connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1) The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2) The cell-to-cell and terminal connections are clean, tight and coated with anti-corrosion material,

3) The resistance of each cell-to-cell* and terminal connection is less than or equal to ~~250 micro-ohms~~ 150 x 10^{-6} ohm, and

4) The battery charger will supply at least 400 amperes at 130 volts for at least 4 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual and/or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test;

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test ~~; and~~ required by Spec 4.8.2.3.2.d.

f. At least once per 18 months during shutdown by giving ~~Annual~~ performance discharge tests of battery capacity ~~shall be given~~ to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

* The resistance of cell-to-cell connecting cables does not have to be included.

TABLE 4.8-3

BATTERY SURVEILLANCE REQUIREMENTS

PARAMETER	CATEGORY A(1)	CATEGORY B(2)	
	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE(3) VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts 6 ~~(c)~~	> 2.07 volts
Specific Gravity 4 ~~(a)~~	≥ 1.195 5 ~~(b)~~	≥ 1.190	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.200	Average of all connected cells ≥ 1.190 5 ~~(b)~~

4 ~~(a)~~ Corrected for electrolyte temperature and level.
5 ~~(b)~~ Or battery charging current is less than ~~5~~2 amps when on charge.
6 ~~(c)~~ Corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.8.3.2 As a minimum, one 125-volt battery bank and an associated full-capacity charger shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With the required battery bank and/or full-capacity charger inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required battery bank and/or full-capacity charger to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE in accordance with Specification 4.8.3.1.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

LIMITING CONDITION~~S~~ FOR OPERATION

3.8.4.1 The thermal overload protection and bypass devices, integral with the motor starter, of each valve listed in Table 3.8-1 shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With one or more of the thermal overload protection and/or bypass devices inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected valves.

SURVEILLANCE REQUIREMENTS

4.8.4.1 The above required thermal overload protection and bypass devices shall be demonstrated OPERABLE:

a. At least once per 18 months, by the performance of a ~~CHANNEL FUNCTIONAL~~ TRIP ACTUATION DEVICE OPERATIONAL TEST of the bypass circuitry for those thermal overload devices which are either:

 1) Continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or

 2) Normally in force during plant operation and bypassed under accident conditions.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of:

 1) All thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years, and

 2) All thermal overload devices which are continuously bypassed, such that each continuously bypassed device is calibrated and each valve is cycled through at least one complete cycle of full travel with the motor-operator when the thermal overload device is OPERABLE and not bypassed at least once per 6 years.

TABLE 3.8-1

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
8107	Charging Isolation	Yes
8105	Charging Pumps' Recirculation	Yes
FCV-430	Component Cooling Heat Exch. Outlet	No
LCV-112B	Volume Control Tank Outlet	Yes
8801A	Boron Injection Tank Outlet	Yes
8803A	Boron Injection Tank Inlet	Yes
8807A	Safety Injection/Charging Suction Crosstie	No
8805A	RWST to Charging Pumps	Yes
FCV-437	Raw Water Supply to Auxiliary Feedpumps	No
FCV-441	SG #4 Feedwater Isolation	Yes
FCV-750	RCP Thermal Barrier CCW Return	Yes
FCV-438	SG #1 Feedwater Isolation	Yes
8923A	SI Pump #1 Suction	No
FCV-38	Aux. FWP Turb. Steam Supply	No
8980	RWST to RHR	No
8974A	SI Pumps' Recirculation	No
8992	Spray Additive Tank Outlet	Yes
8000A	Pressurizer RV isolation	No
FCV-601*	Auxiliary Saltwater Pumps Crosstie	No
8808A	Accumulator No. #1 Isolation	No
8802A	SI Pump #1 to Hot Leg	No
8821A	SI Pump #1 to Cold Legs	No
8808D	Accumulator #4 Isolation	No
8808B	Accumulator #2 Isolation	No
8106	Charging Pumps' Recirculation	Yes
8108	Charging Isolation	Yes
LCV-112C	Volume Control Tank Isolation	Yes

* FCV-601 is common to both units

TABLE 3.8-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
8809A	RHR to Cold Legs	No
8801B	Boron Injection Tank Outlet	Yes
8805B	RWST to Charging Pumps	Yes
8700A	RHR Pump #1 Suction	No
8804A	RHR to Charging Pumps	No
FCV-436	RWSR to Auxiliary Feedpump	No
9001A	Spray Isolation	Yes
FCV-363	RCP CCW Return Isolation	Yes
8835	SI Pumps to RCS Cold Legs	No
8701	RCS to RHR System	No
8100	RCP Seal Water Return	Yes
8803B	Boron Injection Tank Inlet	Yes
FCV-431	CCW Heat Exchanger Outlet	No
FCV-641A	RHR Recirculation	No
8716A	RHR to Hot Legs	No
8000B	Pressurizer RV Isolation	No
FCV-439	S/G #2 Feedwater Isolation	Yes
9003A	RHR to Spray	No
FCV-95	Turb. Feedpump Steam Supply	Yes
8994A	Spray Additive Tank Outlet	Yes
8703	RHR to Hot Legs	No
8104	Emergency Borate	No
8982A	Containment Sump RHR Recirculation	No
LCV-106	S/G ~~SG~~ #1 Aux. Feedwater Supply	No
LCV-107	S/G #2 Aux. Feedwater Supply	No
LCV-108	S/G #3 Aux. Feedwater Supply	No
LCV-109	S/G #4 Aux. Feedwater Supply	No

TABLE 3.8-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
FCV-356	RCP CCW Supply Isolation	Yes
8808C	Accumulator #3 Isolation	No
FCV-641B	RHR Recirculation	No
FCV-355	CCW Header C Isolation	Yes
9001B	Spray Isolation	Yes
8982B	Containment Sump RHR Recirculation	No
9003B	RHR to Spray	No
8976	RWST to SI Pumps	No
8804B	RHR to SI Pumps	No
8802B	SI Pump #2 to Hot Legs	No
8112	RCP Seal Water Return Isolation	Yes
FCV-357	RCP Barrier CCW Return Isolation	Yes
FCV-749	RCP Bearing Cooling H_2O Return Isolation	Yes
8702	RCS to RHR Suction	No
FCV-440	S/G #3 Feedwater Isolation	Yes
8716B	RHR to RCS Hot Legs	No
FCV-37	Aux. Feedpump Steam Supply	No
8821B	SI Pump #2 to Cold Legs	No
8807B	Safety Injection/Charging Suction Crosstie	No
8000C	Pressurizer RV Isolation	No
8994B	Spray Additive Tank Outlet	Yes
FCV-495	ASW Crosstie	No
FCV-496	ASW Crosstie	No
8700B	RHR Pump #2 Suction	No
8974B	SI Pumps Recirculation	No
8809B	RHR to Cold Legs	No
8923B	SI Pump #2 Suction	No

ELECTRICAL POWER SYSTEMS

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 All containment penetration conductor overcurrent protective devices given in Table 3.8-2 shall be OPERABLE.

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

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) given in Table 3.8-2 inoperable:

a. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.2 All containment penetration conductor overcurrent protective devices given in Table 3.8-2 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1) By verifying that the medium voltage 12 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers and performing the following:

a) A CHANNEL CALIBRATION of the associated protective relays,

b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breaker and overcurrent control circuit function as designed, and

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2) By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of "drawout" type circuit breakers shall consist of a CHANNEL CALIBRATION of the associated solid-state trip device for both the long-time delay trip element and the short-time delay element along with a breaker functional test. Testing of molded case circuit breakers shall consist of injecting a current with a value equal to 200% (for D.C. breakers) and 300% (for A.C. breakers) of the pickup of the time delay element and verifying that the circuit breaker operates within the time delay band for that current specified by the manufacturer. The instantaneous element of molded case circuit breakers shall be tested by injecting a current equal to -25%, +40% of the pickup value of the element and verifying that the circuit breaker trips with no intentional time delay. Circuit breakers found out-of-tolerance during functional testing shall be replaced prior to resuming operation. Circuit breakers that fail to trip magnetically before the withstand capability of the penetration conductor is reached shall be declared inoperable. Circuit breakers that fail to trip thermally before the manufacturers maximum tolerance shall be declared inoperable. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested; and

3) By verifying that the thermal overload devices integral with the motor starters, used for penetration overcurrent protection, are OPERABLE by selecting a representative sample of at least 10% of the motor overload devices and performing a CHANNEL CALIBRATION. Motor overloads found inoperable shall be restored to OPERABLE status prior to resuming operation. For each motor overload device found inoperable, a CHANNEL CALIBRATION shall be performed on an additional representative sample of at least 10% of all the motor overload devices of the inoperabe type until no more failures are found or a CHANNEL CALIBRATION has been performed on all motor overload devices of that type.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8-2

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

UNIT 2

PRIMARY DEVICE NO.	BACK-UP DEVICE NO.	EQUIPMENT DESCRIPTION
12 KV		
52VE7	52VE7R	Reactor Coolant Pump 1
52VD3	52VD3R	Reactor Coolant Pump 2
52VE6	52VE6R	Reactor Coolant Pump 3
52VD4	52VD4R	Reactor Coolant Pump 4
480 V "DRAWOUT"		
5223EI	5223EIR	Load Center 3I Feeder
5223DJ	5223DJR	Load Center 3J Feeder
480 V MCC		
522F02*	522F02R	Containment Fan Clr 1
522F01*	522F01R	Containment Fan Clr 2
522G01*	522G01R	Containment Fan Clr 3
522H01*	522H01R	Containment Fan Clr 4
522G02*	522G02R	Containment Fan Clr 5
522F40*	522F40R	PORV 8000A
522G46*	522G46R	PORV 8000B
522H33*	522H33R	PORV 8000C
522F46*	522F46R	Accum Inj to Cold Lp 8808A
522G07*	522G07R	Accum Inj to Cold Lp 8808B
522H14*	522H14R	Accum Inj to Cold Lp 8808C
522G05*	522G05R	Accum Inj to Cold Lp 8808D
522F23	522F23R	FCV750 RCP Barrier CCW Ret
522G38	522G38R	Emergency Ltg Xfmr
522G25*	522G25R	8701 RHR Suction
522H19*	522H19R	8702 RHR Suction
522G56*	522G56R	8703 RHR Recirc
522H18	522H18R	FCV749 RCP Brg Oil Clr
522H27	522H27R	8112 RCP Seal Water Ret
522F45	522F45R	Nuclear Rod Pos Racks
522G67	522G67R	H_2 Recomb
522H35	522H35R	H_2 Recomb
522G37*	522H37R	FCV658 H_2 Recomb Iso
522H73*	522H73R	FCV659 H_2 Recomb Iso
5222J38*	5222J38R	CRDM Exh Fan 1
5222I40*	5222I40R	CRDM Exh Fan 2
5222J39*	5222J39R	CRDM Exh Fan 3
5222I39*	5222I39R	CRDM Exh Fan 4
5222I37*	5222I37R	RCWP Thrust Brg Lift Pump 1

*These primary devices have motor overload relays that are an integral part of the motor starter as part of the primary protective device.

TABLE 3.8-2 (continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NO.	BACK-UP DEVICE NO.	EQUIPMENT DESCRIPTION
5222J13*	5222J13R	RCWP Thrust Brg Lift Pump 2
5222I21*	5222I21R	RCWP Thrust Brg Lift Pump 3
5222J18*	5222J18R	RCWP Thrust Brg Lift Pump 4
5222I10*	5222I10R	Containment Sump Pump 1
5222J10*	5222J10R	Containment Sump Pump 2
5222I30*	5222I30R	Containment Sump Pump 3
5222J29*	5222J29R	Containment Sump Pump 4
5222I31*	5222I31R	Reactor Clnt Drain Pump 1
5222J07*	5222J07R	Reactor Clnt Drain Pump 2
5222I26*	5222I26R	Iodine Rem Exh Fan 5
5222J02*	5222J02R	Iodine Rem Exh Fan 6
5222I25*	5222I25R	Reactor Cavity Sump Pump 1
5222J11*	5222J11R	Reactor Cavity Sump Pump 2
5222I08	5222I08R	Containment Sup Fan 6
480 V HEATER PANELS		
52PH211	52PH211R	Pressurizer Heater Panel 1
52PH212	52PH212R	Pressurizer Heater Panel 1
52PH213	52PH213R	Pressurizer Heater Panel 1
52PH214	52PH214R	Pressurizer Heater Panel 1
52PH215	52PH215R	Pressurizer Heater Panel 1
52PH216	52PH216R	Pressurizer Heater Panel 1
52PH221	52PH221R	Pressurizer Heater Panel 2
52PH222	52PH222R	Pressurizer Heater Panel 2
52PH223	52PH223R	Pressurizer Heater Panel 2
52PH224	52PH224R	Pressurizer Heater Panel 2
52PH225	52PH225R	Pressurizer Heater Panel 2
52PH226	52PH226R	Pressurizer Heater Panel 2
52PH227	52PH227R	Pressurizer Heater Panel 2
52PH231	52PH231R	Pressurizer Heater Panel 3
52PH232	52PH232R	Pressurizer Heater Panel 3
52PH233	52PH233R	Pressurizer Heater Panel 3
52PH234	52PH234R	Pressurizer Heater Panel 3
52PH235	52PH235R	Pressurizer Heater Panel 3
52PH236	52PH236R	Pressurizer Heater Panel 3
52PH237	52PH237R	Pressurizer Heater Panel 3
52PH241	52PH241R	Pressurizer Heater Panel 4
52PH242	52PH242R	Pressurizer Heater Panel 4
52PH243	52PH243R	Pressurizer Heater Panel 4
52PH244	52PH244R	Pressurizer Heater Panel 4
52PH245	52PH245R	Pressurizer Heater Panel 4
52PH246	52PH246R	Pressurizer Heater Panel 4

*These primary devices have motor overload relays that are an integral part of the motor starter as part of the primary protective device.

TABLE 3.8-2 (continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NO.	BACK-UP DEVICE NO.	EQUIPMENT DESCRIPTION
125 VDC		
722514	722514R	Emergency Ltg Panel
120 VAC		
52PHTAC29	52PHTAC29R	Heat Trace 415A
52PHTBC29	52PHTBC29R	Heat Trace 415B
52PY2126	52PY2126R	RCP 1 Leakage Pres Instr
52PY2226	52PY2226R	RCP 2 Leakage Pres Instr
52PY2326	52PY2326R	RCP 3 Leakage Pres Instr
52PY2428	52PY2428R	RCP 4 Leakage Pres Instr
52PY2513	52PY2513R	Sump Pump Evap Panel

*These primary devices have motor overload relays that are an integral part of the motor starter as part of the primary protective device.

FUSE	52PY23A25	MISC.
FUSE	52PY2221	MISC
FUSE	52PY2334	MISC.
FUSE	52PY2422	MISC.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITIONS FOR OPERATION

3.9.1 ~~With the reactor vessel head closure bolts less than fully tensioned or with the head removed,~~ The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met either:

a. ~~Either~~ a K_{eff} of 0.95 or less, which includes a 1% ~~delta~~ Δ k/k conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor ~~pressure~~ vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once each 72 hours.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE each with continuous visual indication in the control room and one with audible indication in the containment and the control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A ANALOG CHANNEL ~~FUNCTIONAL~~ OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A ANALOG CHANNEL ~~FUNCTIONAL~~ OPERATIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITIONS FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor ~~pressure~~ vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor ~~pressure~~ vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor ~~pressure~~ vessel.

REFUELING OPERATIONS

3/4.9.4 CONTAINMENT PENETRATIONS

LIMITING CONDITION~~S~~ FOR OPERATION

3.9.4 The containment penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1) Closed by an isolation valve, blind flange, or manual valve, or

2) Be capable of being closed by an OPERABLE automatic containment ventilation isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within ~~the~~ containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in ~~the~~ containment ~~building~~.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment ventilation isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in ~~the~~ containment by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment ventilation isolation valves per Specification 4.6.3.2c.

REFUELING OPERATIONS

3/4.9.5 COMMUNICATIONS

LIMITING CONDITIONS FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within ~~one~~ hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

REFUELING OPERATIONS

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITIONS FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

1) A minimum capacity of 3250 pounds, and

2) An overload cut off limit less than or equal to 2700 pounds.

b. The auxiliary hoist used for movement of control rods having:

1) A minimum capacity of 700 pounds, and

2) A load indicator which shall be monitored to preclude lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor ~~pressure~~ vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor ~~pressure~~ vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor ~~pressure~~ vessel shall be demonstrated OPERABLE within 100 hours prior ~~to the start of such operations~~ to removal of the reactor vessel head by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2700 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor ~~pressure~~ vessel shall be demonstrated OPERABLE within 100 hours prior ~~to the start of such operations~~ to removal of the reactor vessel head by performing a load test of at least 700 pounds.

REFUELING OPERATIONS

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING

LIMITING CONDITIONS FOR OPERATION

3.9.7 Loads in excess of 2500 pounds* shall be prohibited from travel over fuel assemblies in the spent fuel pool.**

APPLICABILITY: With fuel assemblies in the spent fuel pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Loads shall be verified to be less than 2500 pounds prior to movement over fuel assemblies in the spent fuel pool.

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*The movable fuel handling building walls may travel over fuel assemblies in the spent fuel pool.

~~**During the period of January 6, 1983 through June 20, 1983, relief from this specification is granted to permit the installation and removal of temporary spent fuel pool steel plate covers. During this period, loads in excess of 500 pounds shall be prohibited from travel over the spent fuel steel plate covers.~~

(3)

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITIONS FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) trains shall be OPERABLE and at least one RHR train shall be in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor ~~pressure~~ vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR trains OPERABLE, immediately initiate corrective action to return the required RHR trains to OPERABLE status, or to establish at least 23 feet of water above the reactor ~~pressure~~ vessel flange, as soon as possible.

b. With no RHR train 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 RHR train to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIRMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant ~~The required residual heat removal trains shall be determined OPERABLE per Specification 4.0.5.~~ at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

* Prior to initial criticality, the RHR train may be removed from operation for up to 1 hour per 8-hour period during performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.9.9 The Containment Ventilation Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the Containment Ventilation Isolation System inoperable, close each of the ventilation penetrations providing direct access from the containment atmosphere to the outside atmosphere. ~~The provisions of Specification 3.0.4 are not applicable~~.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Ventilation Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment ventilation isolation occurs on a high radiation test signal from the plant vent noble gas activity monitoring instrumentation channels.

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITIONS FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor ~~pressure~~ vessel flange.

APPLICABILITY: During movement of fuel assemblies either or control rods within the containment ~~reactor pressure vessel while in MODE 6~~ when the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the ~~pressure~~ vessel.
reactor

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods. ~~within the reactor pressure vessel~~.

REFUELING OPERATIONS

3/4.9.11 WATER LEVEL-SPENT FUEL POOL

LIMITING CONDITION~~S~~ FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

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SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool.

3/4.9.12 FUEL HANDLING BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two Fuel Handling Building Ventilation Systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the spent fuel pool.

ACTION:

a. With one Fuel Handling Building Ventilation System inoperable, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the OPERABLE Fuel Handling Building Ventilation System is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no Fuel Handling Building Ventilation System OPERABLE, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool until at least one Fuel Handling Building Ventilation System is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Handling Building Ventilation Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1) Visually verifying that with the system operating at a flow rate of 35,750 cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, that the damper valve M-29 is closed;

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedures guidance in 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 35,750 cfm ± 10%:

3) 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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 4.3%; and

4) Verifying a system flow rate of 35,750 cfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975~~ 80

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 Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 4.3%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4.1 inches Water Gauge while operating the system at a flow rate of 35,750 cfm ± 10%.

2) Verifying that on a High Radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks, and

3) Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-~~1975~~ for a DOP test aerosol while oprating the system at a flow rate of 35,750 cfm ± 10%; and

8c

SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-19~~75~~ for a halogenated hydrocarbon test gas while operating the system at a flow rate of 35,750 cfm ± 10%.

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REFUELING OPERATIONS

3/4.9.13 SPENT FUEL SHIPPING CASK MOVEMENT

LIMITING CONDITIONS FOR OPERATION

3.9.13 No spent fuel shipping cask handling operation near the spent fuel pool (i.e., any movement of a cask located north of column line 12.9 for Unit 1) shall be performed unless spent fuel in all locations in Racks 5 and 6 ~~(as shown in Figure 9.1-46 of the FSAR)~~ has decayed for at least 1000 hours since shutdown.

APPLICABILITY: During all cask handling operations.

ACTION:

With the requirements of the above specification not satisfied, move the cask out of the specified area(s), or move spent fuel which has decayed less than 1000 hours from all locations in Racks 5 and 6.

or south of column line 23.1 for Unit 2

SURVEILLANCE REQUIREMENTS

4.9.13 The decay time of the fuel in Racks 5 and 6 shall be verified to be at least 1000 hours prior to the movement of the cask into the specified area.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION~~S~~ FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITIONS FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be ~~≤~~ less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed Specifications ~~below~~ shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specification 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITIONS FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 531°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 531°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to an ANALOG CHANNEL ~~FUNCTIONAL~~ OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 531°F at least once per 30 minutes during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITIONS FOR OPERATION

3.10.4 The limitations of the following requirements may be suspended:

a. Specification 3.4.1.1 - During the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

1) The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

2) The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

b. Specification 3.4.1.2 - During the performance of natural circulation tests in MODE 3 provided at least three reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

c. Specification 3.7.1.3 - During the performance of natural circulation tests in MODE 3 provided the fire water tank and its flow path to the auxiliary feedwater pump is OPERABLE as stated in Specification 4.7.1.3.2.

APPLICABILITY: During operation below the P-7 Interlock Setpoint or performance of natural circulation tests.

ACTION:

a. With the THERMAL POWER greater than the P-7 Interlock Setpoint during the performance of startup and PHYSICS TESTS, immediately open the Reactor trip breakers.

b. With less than the above required reactor coolant loops OPERABLE during the performance of the natural circulation tests, immediately place two reactor coolant loops in operation.

c. With the condensate storage tank inoperable, take the appropriate ACTION statement given in Specification 3.7.1.3.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than the P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

4.10.4.3 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to the initiation of the natural circulation tests and at least once per 4 hours during the natural circulation tests by verifying correct breaker alignments and indicated power availability.

4.10.4.4 The fire water tank and its flow path to the auxiliary feedwater pump shall be determined OPERABLE within 4 hours prior to the initiation of the natural circulation tests and at least once per 4 hours during the natural circulation tests in accordance with Specification 4.7.1.3.2.

SPECIAL TEST EXCEPTIONS

3/4.10.4 POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.10.4 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time; ~~and~~

~~b. The rod position indicator is OPERABLE during the withdrawal of the rods.*~~

3

APPLICABILITY: MODES 3, 4 and 5 during performance of rod drop time measurements and during surveillance of digital rod position indication for OPERABILITY.

ACTION:

With the position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4 The above required rod position indication systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during the rod drop time measurements by verifying the demand position Indication System and the Digital Rod Position Indication Systems agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

~~*This requirement is not applicable during the initial calibration of the rod position indication system provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one control rod bank is withdrawn from the fully inserted position at one time.~~

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCP to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

Insert

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

in liquid effluents to UNRESTRICTED AR.

3.11.1.1 The concentration of radioactive material released ~~from the site~~ (see Figure 5.1-3) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microCuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

in liquid effluents to UNRESTRICTED AREA:

With the concentration of radioactive material released ~~from the site~~ exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

~~4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11-1. The results of pre-release analyses shall be used with the calculational methods in the ODCP to assure that the concentration at the point of release is maintained within the limits of Specification 3.11.1.1.~~

INSERT

~~4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11-1. The results of the previous post-release analyses shall be used with the calculational methods in the ODCP to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.~~

~~4.11.1.1.3 The radioactivity concentration of liquids discharged from continuous release points shall be determined by collection and analysis of samples in accordance with Table 4.11-1. The results of the analyses shall be used with the calculational methods in the ODCP to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.~~

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)(1)
1. Batch Waste Release Tanks(4)	P Each Batch	P Each Batch	Principal Gamma Emitters(6)	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	1×10^{-5}
	P Each Batch	M Composite(2)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
			~~P-32~~	~~1×10^{-6}~~
	P Each Batch	Q Composite(2)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
2. Continuous Releases(5) a. Steam Generator Blowdown Tank	D Grab Sample	W Composite(3)	Principal Gamma Emitters(6)	5×10^{-7}
			I-131	1×10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	D Grab Sample	M Composite(3)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
			~~P-32~~	~~1×10^{-6}~~
	D Grab Sample	Q Composite(3)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
b. Oily Water Separator Effluent	D Grab Sample	W Composite(3)	Principal ~~Gross~~ Gamma Emitters(6)	5 ~~1~~ $\times10^{-7}$

(1)

(3)

TABLE 4.11-1 (Continued)

that will yield a net count, above system background,

TABLE NOTATION

defined, for purposes of these specifications, as

~~a.~~ (1) The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$\text{LLD} = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD = ~~is~~ the "a priori" lower limit of detection ~~as defined above~~ (~~as~~ microcuries per unit mass or volume),

s_b = ~~is~~ the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (~~as~~ counts per minute),

E = ~~is~~ the counting efficiency (~~as~~ counts per ~~transformation~~ disintegration),

V = ~~is~~ the sample size (~~in~~ units of mass or volume),

2.22×10^6 = ~~is~~ the number of ~~transformations~~ disintegrations per minute per microcurie,

Y = ~~is~~ the fractional radiochemical yield (when applicable),

λ = ~~is~~ the radioactive decay constant for the particular radionuclide (sec⁻¹), ~~(…)~~ and

Δt = ~~is~~ the elapsed time between the midpoint of sample collection and the time of counting ~~(for plant effluents, not environmental samples)~~ (sec) ~~(…)~~.

~~The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance.~~ Typical values of E, V, Y, and Δt shall be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

(1

TABLE 4.11-1 (Continued)

TABLE NOTATION

~~b.~~ (2) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

~~c.~~ (3) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be composited in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be throughly mixed in order for the composite sample to be representative of the effluent release.

~~d.~~ (4) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed, by a method described in the ~~plant manual~~ ODCF, to assure representative sampling.

~~e.~~ (5) A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of system that has an input flow during the continuous release.

~~f.~~ (6) The principal gamma emitters for which the LLD specification applies include ~~exclusively are~~ the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, ~~and~~ Ce-144. and This list does not mean that only these nuclides are to be ~~detected and reported~~ considered. Other gamma peaks ~~which~~ that are ~~measurable and~~ identifiable, together with the above nuclides, shall also be ~~identified~~ analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.? in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

shall also be measured but with an LLD of 5×10^{-6}

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

a MEMBER OF THE PUBLIC

3.11.1.2 The dose or dose commitment to ~~an individual~~ from radioactive materials in liquid effluents released, from each ~~reactor~~ unit, ~~from the site~~ to UNRESTRICTED AREAS (see Figure 5.1-3) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the whole ~~total~~ body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the whole ~~total~~ body and to less than or equal to 10 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, ~~in lieu of any other report required by Specification 6.9.1~~, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report ~~which~~ that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions ~~to be~~ that be taken to reduce the releases ~~of radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the subsequent three calendar quarters, so that the cumulative dose or dose commitment to an individual from these releases is within 3 mrem to the total body and 10 mrem to any organ~~. and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 ~~Dose Calculations.~~ Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCP at least once per 31 days.

6

RADIOACTIVE EFFLUENTS

LIQUID RAD WASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System* shall be OPERABLE and ~~The~~ appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS ~~the site~~ (see Figure 5.1-3) ~~when averaged over 31 days~~, would exceed 0.06 mrem to the ~~total~~ whole body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY: At all times.

ACTION:

a. With ~~the~~ any portion of the Liquid Radwaste Treatment System not in operation and ~~inoperable for more than 31 days or~~ with radioactive liquid waste being discharged without treatment and in excess of the above limits, ~~in lieu of any other report required by Specification 6.9.1~~, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. ~~Identification of the inoperable equipment or subsystems and the reason for inoperability,~~ Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability.

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days, in accordance with the ~~ODCP~~ methodology and parameters in the ODCP when Liquid Radwaste Treatment Systems are not being fully utilized.

4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be ~~demonstrated~~ considered OPERABLE by meeting ~~operating the Liquid Radwaste System equipment for at least 10 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquid effluents during the previous 92 days.~~ Specifications 3.11.1.1 and 3.11.1.2.

~~*Per reactor unit.~~

*The Liquid Radwaste Treatment SYSTEM IS common To both Units.

1

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any temporary outdoor tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the temporary outdoor tanks 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 this condition in the next Semiannual Radioactive Release Report, pursuant to Specification 6.9.1.6

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each ~~of the above listed~~ temporary outdoor tank~~s~~ shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

3

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate ~~in unrestricted areas~~ due to radioactive materials released in gaseous effluents from the site (see Figure 5.1-3) shall be limited to areas at or beyond the SITE BOUNDARY to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the ~~total~~ whole body and less than or equal to 3000 mrem/yr to the skin, and

b. For ~~all radioiodines~~ Iodine 131, for Iodine 133, for tritium, and for all ~~radioactive materials~~ radionuclides in particulate form ~~and radionuclides (other than noble gases)~~ with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the ~~methods~~ methodology and procedures of the ODCP.

4.11.2.1.2 The dose rate due to ~~radioactive materials, other than noble gases,~~ Iodine 131, Iodine 133, Tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodologies and procedures of the ODCP by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) ($\mu Ci/ml$)[a(1)]
A.[1] Waste Gas Decay Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g(7)]	$1x10^{-4}$
B.[2] Containment Purge	P Each Purge[b(2)] Grab Sample	P Each Purge[b(2)]	Principal Gamma Emitters[g(7)]	$1x10^{-4}$
			H-3	$1x10^{-6}$
C.[3] Plant Vent	M[b(2)] Grab Sample	M[b(2)]	Principal Gamma Emitters[g(7)]	$1x10^{-4}$
	W[c,e(3)(5)] Grab Sample	W	H-3	$1x10^{-6}$
D.[4] All Release Types as listed in A,[1] B,[2] C[3] above at the plant vent.	Continuous[f(6)]	W[d(4)] Charcoal Sample	I-131	$1x10^{-12}$
			I-133	$1x10^{-10}$
	Continuous[f(6)]	W[d(4)] Particulate Sample	Principal Gamma Emitters[g(7)] (I-131, Others)	$1x10^{-11}$
	Continuous[f(6)]	M Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous[f(6)]	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$
E.[5] Steam Generator Blowdown Tank Vent	M[b(8)]	M[b(8)]	Principal Gamma Emitters[g(7)]	$1 x 10^{-4}$

TABLE 4.11-2 (Continued)

TABLE NOTATION

(1) ~~a.~~ The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above sytem background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, ~~(~~which may include radiochemical separation~~)~~:

$$\text{LLD} = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD ~~is~~ = the "a priori" lower limit of detection ~~as defined above~~ (~~as~~ microcurie per unit mass or volume),

s_b ~~is~~ = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (~~as~~ counts per minute),

E ~~is~~ = the counting efficiency (~~as~~ counts per ~~transformation~~ disintegration),

V ~~is~~ = the sample size (~~in~~ units of mass or volume),

2.22×10^6 is the number of ~~transformations~~ disintegrations per minute per microcurie,

Y ~~is~~ = the fractional radiochemical yield ~~(~~when applicable~~)~~,

λ ~~is~~ = the radioactive decay constant for the particular radionuclide (sec[-1]) ~~as~~

Δt ~~is~~ = the elapsed time between the midpoint of sample collection and time of counting (sec) ~~(for plant effluents, not environmental samples)~~.

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. Typical values of E, V, Y, and Δt shall be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

TABLE 4.11-2 (Continued)

TABLE NOTATION

(2) ~~b.~~ Sampling and Analyses shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% ~~percent~~ of the RATED THERMAL POWER within a ~~one~~ hour period.

(3) ~~c.~~ Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

(4) ~~d.~~ Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% ~~percent~~ of RATED THERMAL POWER ~~in one~~ within 1 hour period and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the Dose Equivalent I-131 concentration in the reactor coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

(5) ~~e.~~ Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

(6) ~~f.~~ The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

(7) ~~g.~~ The principal gamma emitters for which the LLD specification applies include ~~exclusively are~~ the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 ~~for gaseous emissions~~ in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 ~~for~~ in Iodine and particulate ~~emissions~~ releases. This list does not mean that only these nuclides are to be ~~detected and reported~~ considered. Other gamma peaks ~~which~~ that are ~~measureable and~~ identifiable, together with those of the above nuclides, shall also be ~~identified~~ analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.6 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(8) ~~h.~~ Grab samples shall be taken and analyzed at least once per 31 days whenever there is flow through the steam generator blowdown tank. Releases of radioiodines shall be estimated based on secondary coolant concentration and partitioning factors during releases or shall be measured.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, frc each ~~reactor~~ unit, ~~from the site~~ to areas at or beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gam radiation and less than or equal to 10 mrad for beta radiation and

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseou effluents exceeding any of the above limits, ~~in lieu of any other report required by Specification 6.9.1~~, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Spec Report which identifies the cause(s) for exceeding the limit(s) anc defines the corrective actions that have been ~~to be~~ taken to reduce the releases ~~of radioactive noble gases in gaseous effluents during the remainder of the current calendar quarter and during the subsequent three calenc quarters, so that the cumulative dose is within 10 mrad for gamma radiation and 20 mrad for beta radiation~~. and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable

SURVEILLANCE REQUIREMENTS

4.11.2.2 ~~Dose Calculations~~ Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with methodology and parameters in the ODCP at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND ~~RADIOIODINES~~, RADIOACTIVE MATERIALS IN PARTICULATE FORM, ~~AND RADIONUCLIDE OTHER THAN NOBLE GASES~~

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, Tritium and radionuclides ~~an individual from radioiodines and radioactive materials~~ in particulate form, ~~and radionuclides (other than noble gases)~~ with half-lives greater than 8 days in gaseous effluents released, from each ~~reactor~~ unit, ~~from~~ to ~~the site~~ areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ and,

b. During any calendar year: Less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131, Iodine-133, Tritium, radionuclides ~~radioiodines, radioactive materials~~ in particulate form, ~~or radionuclides (other than noble gases)~~ with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, ~~in lieu of any other report required by Specification 6.9.1~~, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been ~~to be~~ taken to reduce the releases ~~of radio-~~ iodines and radioactive materials in particulate form, and radionuclides (other than nobles gases) with half-lives greater than 8 days in gaseous effluents during the remainder of the current calendar quarter and during the subsequent three calendar quarters, so that the cumulative dose or dose commitment to an individual from these releases is within 15 mrem to any organ and the proposed actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 ~~Dose Calculations~~ Cumulative dose contributions for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days for the current calendar quarter and current calendar year shall be determined in accordance with the methodology and parameters in the ODCP at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of ~~the GASEOUS RADWASTE SYSTEM~~ these systems shall be used to reduce releases of radioactivity ~~radioactive materials in gaseous waste prior to their discharge~~ when the projected ~~gaseous effluent air~~ doses in 31 days due to gaseous effluent releases from each unit to areas at or beyond the SITE BOUNDARY ~~from the site~~ (see Figure 5.1-3), ~~when averaged over 31 days~~, would exceed 0.2 mrad to air from ~~for~~ gamma radiation ~~and~~ or 0.4 mrad to air from ~~for~~ beta radiation, ~~The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site (see Figure 5.1-3) when averaged over 31 days would exceed~~ or 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive ~~the GASEOUS RADWASTE SYSTEM and/or the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days or with~~ gaseous waste being discharged without treatment and in excess of the above limits, ~~in lieu of any other report required by Specification 6.9.1~~, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report ~~which~~ that includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from ~~the site~~ each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days, in accordance with the ~~ODCM~~ methodology and parameters in the ODCM when Gaseous Radwaste Treatment systems are not being fully utilized.

4.11.2.4.2 The installed GASEOUS RADWASTE SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM shall be ~~demonstrated~~ considered OPERABLE by ~~operating the GASEOUS RADWASTE SYSTEM equipment and VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 10 minutes~~, at least once per 92 days ~~unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days~~. meeting Specifications 3.11.2.1 and 3.11.2.2 or 3.11.2.3

~~*Per reactor unit.~~

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the GASEOUS RADWASTE SYSTEM shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the GASEOUS RADWASTE SYSTEM greater than 2% by volume but less than or equal 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the GASEOUS RADWASTE SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume ~~within one hour~~ ~~and 2% by volume within 48 hours after initially exceeding 2% by volume~~. then take ACTION a, above

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen* and oxygen in the GASEOUS RADWASTE SYSTEM shall be determined to be within the above limits by ~~continuously~~ monitoring the waste gases in the GASEOUS RADWASTE SYSTEM with the hydrogen and continuous oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

* if monitoring of the waste gases for hydrogen is not performed, the hydrogen concentration shall be assumed to be greater than 4% by volume.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each Gas Decay Tank shall be limited to less than or equal to 10^5 curies noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any Gas Decay 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 this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.c

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each Gas Decay Tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

3/.11.3 SOLID RADIOACTIVE WASTES

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be solidified or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION: Insert

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures and/or the Solid Waste System as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and take appropriate administrative action to prevent recurrence.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM:

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM;

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste; and

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

(1)

RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be OPERABLE and used, as applicable in accordance with a PROCESS CONTROL PROGRAM, for the SOLIDIFICATION and packaging of radioactive wastes to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71 prior to shipment of radioactive wastes from the site.

APPLICABILITY: At all times.

ACTION:

a. With the packaging requirements of 10 CFR Part 20 and/or 10 CFR Part 71 not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

b. With the solid radwaste system inoperable for more than 31 days, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

INSERT

1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

2. Action(s) taken to restore the inoperable equipment to OPERABLE status,

3. A description of the alternative used for SOLIDIFCATION and packaging of radioactive wastes, and

4. Summary description of action(s) taken to prevent a recurrence.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3.1 The solid radwaste system shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION to be performed by a contractor in accordance with a PROCESS CONTROL PROGRAM.

SURVEILLANCE REQUIREMENTS (Continued)

4.11.3.2 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch (300 cu. ft. ± 20%) of the same type of wet radioactive waste.

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations shall be made including direct radiation contributions from the units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete. Insert

6

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

annual (calendar year)

3.11.4 The dose or dose commitment to any ~~member of the public~~, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the ~~total~~ whole body or any organ, (except the thyroid, which shall be limited to less than or equal to 75 mrem) ~~over 12 consecutive months~~.

APPLICABILITY: At all times.

ACTION:

a. ~~With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days, which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.11.4. This Special Report shall include an analysis which estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources (including all effluent pathways and direct radiation) for a 12 consecutive month period that includes the release(s) covered by this report. If the estimated dose(s) exceeds the limits of Specification 3.11.4, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190 and including the specified information of § 190.11(b). Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete. The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the requirements for dose limitation of 10 CFR Part 20, as addressed in other sections of this technical specification.~~

INSERT

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 ~~Dose Calculations~~ Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the ~~ODCP~~ methodology and parameters in the OPCF.

4.11.4.2 Cumulative dose contributions from direct radiation from the units and from outside storage tanks shall be determined in accordance with the methodology and the parameters in the OPCF. This requirement is applicable only under conditions set forth in ACTION a. of Specification 3.11.4.

6

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, ~~in lieu of any other report required by Specification 6.9.1~~, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.6, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, ~~in lieu of any other report required by Specification 6.9.1~~, prepare and submit to the Commission within 30 days, ~~from the end of the affected calendar quarter a Report~~ pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, or 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\cancel{\text{limit}}\ \text{reporting level (1)}} + \frac{\text{concentration (2)}}{\cancel{\text{limit}}\ \text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to ~~an individual~~ A MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 ~~and~~ or 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.6.

c. With milk or ~~fresh leafy vegetable~~ vegetation samples not being performed as ~~unavailable from one or more of the sample locations~~ required by Table 3.12-1, ~~in lieu of any other report required by Specification 6.9.1~~, ~~prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause of the unavailability of samples and~~ identify ~~identifies~~ locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program in the ECDM. The specific locations from which samples were unavailable may then be deleted from the monitoring program ~~those required by Table 3.12-1, provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations.~~ (See next page)

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

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* The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

(1.

RADIOLOGICAL ENVIRONMENTAL MONITORING

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure in the ERMP and shall be analyzed pursuant to the requirements of Tables 3.12-1 and ~~4.12-1~~. the detection capabilities required by Table 4.12-1.

c. (continued)

Pursuant to Specification 6.9.1.4, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ERMP including a revised figure(s) and table for the ERMP reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

See revised Table 3.12-1

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES AND SAMPLE LOCATIONS**	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. AIRBORNE			
Radioiodine and Particulates	$\geq$ 4 stations	Continuous operation of sampler with sample collection as required by dust loading but at least once per 7 days.	Radioiodine canister. Analyze at least once per 7 days for I-131. Particulate sampler. Analyze for gross beta radioactivity $\geq$ 24 hours following filter change. Perform gamma isotopic analysis on each sample when gross beta activity is > 10 times the yearly mean of control samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
2. DIRECT RADIATION	$\geq$ 30 stations, $\geq$ 2 dosimeters at each location.	At least once per 31 days.*	Gamma dose. At least once per 31 days.*

* Except for one station which is inaccessable and sampled and analyzed at least once per 92 days.

** Sample locations are shown on the figure in the ERMP Procedure No. A-8

See revised Table 3.12-1

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES AND SAMPLE LOCATIONS**	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. WATERBORNE			
a. Outfall	1 station	Grab sample collected at least every 31 days and composited at least every 92 days.	Gamma isotopic analysis and tritium analysis at least every 92 days.
b. Drinking	1 station	Grab sample collected at least every 31 days.	I-131 analysis, gross beta and gamma isotopic analysis and tritium analysis at least once per 31 days.
c. Diablo Canyon Creek	1 station	Grab sample collected at least every 31 days and composited at least every 92 days.	Gamma isotopic analysis and tritium analysis at least every 92 days.
4. INGESTION			
a. Milk	≥ 2 stations	At least once per 31 days.	Gamma isotopic and I-131 analysis.
b. Fish and Invertebrates	2 stations	One sample in season, or at least once per 184 days if not seasonal.	Gamma isotopic analysis on edible portions.
c. Food Products	≥ 2 stations	At least once per 31 days, when available.	Gamma isotopic analysis on edible portion.

*Except for one station which is inaccessible ~~and sampled and analyzed at least once per 92 days~~.
**Sample locations are shown on the figure in the ERMP Procedure No. A-8.

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TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation(2)	Thirty-one routine monitoring stations* either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: An inner ring of stations, one in each terrestrial meteorological sector in the general area of the SITE BOUNDARY; An outer ring of stations, one in each terrestrial meteorological sector in the 2.5 to 12 km range from the site; and The balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.	Quarterly.	Gamma dose quarterly

*Inner ring stations: 0S1, WN1, 0S2, 1S1, 2S1, 3S1, 4S1, 5S1, 5S3, 6S1, 7S1, 8S1, 8S2, 9S1, and MT1.
Outer ring stations: 1A1, 0B1, 1C1, 2D1, 3D1, 4C1, 5C1, 6D1, and 7C1
Special interest stations: 4D1, 5F1, 7F1, 7D1, 7D2, and 7C2
Control station: 2F2

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne			
Radioiodine and Particulates	Samples from six locations (MT1, OS2, 8S1, 2F2, 5F1 and 7D1);	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Canister: I-131 analysis weekly.
	Three samples (MT1, OS2 and 8S1) from close to the three SITE BOUNDARY, locations, in different sectors, two in sectors with the highest calculated annual average ground-level D/Q, the other in the South sector;		Particulate Sampler: Gross beta radioactivity analysis following filter change;[3] and gamma isotopic analysis[4] of composite (by location) quarterly.
	One sample (7D1) from the vicinity of a community having the highest calculated annual average ground-level D/Q;		
	One sample (5F1) from the San Luis Obispo area; and		
	One sample (2F2) from a control location.		

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne			
a. Surface	One sample from Diablo Creek (5S2).	Monthly.	Gamma isotopic analysis(4) monthly. Composite for tritium analysis quarterly.
b. Drinking	One sample of plant drinking water (DW1)	Monthly grab sample.	Gamma isotopic analyses(4) monthly. Composite for tritium analysis quarterly.
4. Ingestion			
a. Milk	Samples from milking animals in three locations within 5 km distance having the highest dose potential. If there are none, then one sample from milking animals in each of three areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr. One sample from milking animals at a control location 15 to 30 km distant and in the least prevalent wind direction.	Semimonthly when animals are on pasture; monthly at other times.	Gamma isotopic(4) and I-131 analysis semimonthly when animals are on pasture; monthly at other times.

TABLE 3.12-1 (Continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (Cont'd)			
b. Fish and Invertebrates	One sample rockfish (Sebastes sp.), one sample surfperch (Family Embiotocidae), and one sample abalone (Haliotis sp.), from the vicinity of plant discharge area (DCM).	Sample in season, or semiannually if they are not seasonal.	Gamma isotopic analysis(4) on edible portions of each sample.
	One sample of the above in areas not influenced by plant discharge (POS or PON)		
c. Food Products	Samples of three different kinds of broad leaf vegetation* grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q if milk sampling is not performed.	Monthly during growing season.	Gamma isotopic(4) and I-131 analysis.
	One sample of each of the similar broad leaf vegetation grown 15 to 30 km distant in the least prevalent wind direction if milk sampling is not performed.	Monthly during growing season.	Gamma isotopic(4) and I-131 analysis.

*If broadleaf vegetation is unavailable, other vegetation approximating broadleaf as described in the ERMP will be sampled or additional air sampling will be done in the SE and NNW sectors.

TABLE 3.12-1 (Continued)

TABLE NOTATIONS

(1) Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12-1 in a table and figure(s) in the ERMP. Station codes given in Table 3.12-1 correspond to sample locations given in the ERMP. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment. If specimens are unobtainable due to sampling equipment malfunction, effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable specific alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ERMP including a revised figure(s) and table for the ERMP reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples for that pathway and justifying the selection of the new location(s) for obtaining samples.

(2) For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation.

(3) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

(4) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m³)	FISH (pCi/Kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/Kg, wet)
H-3	~~2×10^4~~ (*a) 20,000				
Mn-54	~~1×10^3~~ 1000		~~3×10^4~~ 30,000		
Fe-59	~~4×10^2~~ 400		~~1×10^4~~ 10,000		
Co-58	~~1×10^3~~ 1000		~~3×10^4~~ 30,000		
Co-60	~~3×10^2~~ 300		~~1×10^4~~ 10,000		
Zn-65	~~3×10^2~~ 300		~~2×10^4~~ 20,000		
Zr-Nb-95	~~4×10^2~~ 400				
I-131	2 **	0.9		3	~~1×10^2~~ 100
Cs-134	30	10	~~1×10^3~~ 1000	60	~~1×10^3~~ 1000
Cs-137	50	20	~~2×10^3~~ 2000	70	~~2×10^3~~ 2000
Ba-La-140	~~2×10^2~~ 200			~~3×10^2~~ 300	

(*a) For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/l may be used.

** If no drinking water pathway exists, a value of 20 pCi/l may be used.

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[1][2]

~~MAXIMUM VALUES FOR THE~~ LOWER LIMITS OF DETECTION (LLD)[a,d]

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
gross beta	4	~~1×10^{-2}~~ 0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
~~Zr-95~~	~~30~~ c					
Zr-Nb-95	15					
I-131	1 ~~B~~ **	~~7×10^{-2}~~ 0.07		1	60	
Cs-134	15	~~5×10^{-2}~~ 0.05	130	15	60	150
Cs-137	18	~~6×10^{-2}~~ 0.06	150	18	80	180
~~Ba-140~~	~~60~~ e			60		
Ba-La-140	15			15		

* For surface water samples, a value of 3000 pCi/l may be used.

** If no drinking water pathway exists, a value of 15 pCi/l may be used.

(i).

(1) } on page 3/4 12-8
(2) }

TABLE 4.12-1 (Continued)

TABLE NOTATION

defined, for purposes of these specifications, as

(j)a. The LLD is the smallest concentration of radioactive material in a sample that will yield a counts per sample gross that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD ~~is~~ = the "a priori" lower limit of detection ~~as defined above (as~~ (picocurie per unit mass or volume),

s_b ~~is~~ = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (~~as~~ counts per minute),

E ~~is~~ = the counting efficiency (~~as~~ counts per ~~transformation~~ disintegration),

V ~~is~~ = the sample size (~~in~~ units of mass or volume),

2.22 ~~is~~ = the number of ~~transformation~~ disintegrations per minute per picocurie,

Y ~~is~~ = the fractional radiochemical yield, (when applicable),

λ ~~is~~ = the radioactive decay constant for the particular radionuclide (sec⁻¹), and

Δt ~~is~~ = the elapsed time between environmental ~~sample~~ collection, (or end of the sample collection period), and time of counting (sec) ~~(for environmental samples, not plant effluent samples).~~

~~The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples).~~ Typical values of E, V, Y and Δt shall be used in the calculation.

INSERT

It should be recognized that the LLD is defined as an _a priori_ (before the fact) limit representing the capability of a measurement system and not as an _a posteriori_ (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall

- next page -

(1)

TABLE 4.12-1 (Continued)

TABLE NOTATION

~~W.B. LLD for drinking water. For surface water samples, the LLD of gamma isotopic analysis may be used.~~

This list does not mean that only these nuclides are to be considered.

(1) ~~c.~~ Other peaks ~~which are measurable and~~ that are identifiable, together with those of the above ~~radio~~nuclides ~~in Table 4.12-1,~~ shall also be ~~identified~~ analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

3. (continued)
be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13, Revision 1, Jul, 1977.

6

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

broad leaf vegetation

50m²

within a distance of 8km (5 miles)

3.12.2 A Land Use Census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 ~~square feet~~ producing ~~fresh leafy vegetables~~ in each of the 16 meteorological sectors ~~within a distance of five miles~~.

5

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) ~~which~~ that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, ~~in lieu of any other report required by Specification 6.9.1., prepare and submit to the Commission within 30 days,~~ pursuant to Specification 6.9.1.6, ~~a Special Report which~~ identify the new location(s) in the next Semiannual Radioactive Effluent Release Report.

b. With a Land Use Census identifying a location(s) ~~which~~ that yields a calculated dose or dose commitment (via the same exposure pathway) 20 ~~percent~~ % greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, ~~in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location.~~ add the new location(s) within 3 ~~shall be added~~ to the Radiological Environmental Monitoring Program given in the ~~within 30 days~~. The sampling location(s) excluding the control station location, having the lowest calculated dose or dose commitment(s) (via the same exposure pathway), may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted. Pursuant to Specification 6.9.1, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the EEMP including a revised figure(s) and table(s) for the EEMP reflecting the new location(s) with information supporting the change in sampling locations.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

during the growing season

4.12.2 The land use census shall be conducted at least once per 12 months ~~between the dates of (June 1 and October 1)~~ using that information ~~which~~ that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the Land Use Census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

---

of at least three different kinds of vegetation

*Broad leaf vegetation sampling may be performed at the ~~site boundary~~ in ~~the~~ each of two direction sectors with the highest D/Q in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table ~~3.12-1~~ 3.12, Part 4.c shall be followed, including analysis of control samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on all radioactive materials, supplied as part of an Interlaboratory Comparison Program ~~which~~ the has been approved by the Commission, that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report, pursuant to Specification 6.9.1.1.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

The Interlaboratory Comparison Program shall be described in the ERMP.

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program ~~and in accordance with the ERMP (or participants in the EPA crosscheck program shall provide the EPA program code designation for the unit)~~ shall be included in the Annual Radiological Environmental Operating Report, pursuant to Specification 6.9.1.1.

BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The summary statements contained in this section provide the bases for the specifications in Sections 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

3/4.0 APPLICABILITY

BASES

(3)

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4. In the event of a disagreement between the requiremen[ts] in these Technical Specifications and those stated in an applicable Federal Regulation or A[ct] the requirements stated in the applicable Federal Regulation or Act shall take preceden[ce] and shall be met except where a specific exemption to the Regulation has been granted [by] the Technical Specifications or Facility License.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Conditions for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION Statements and whose occurrence would violate the intent of a specification. For example, Specification 3.5.2 requires two ~~independent~~ ECCS Subsystems to be OPERABLE and provides explicit ACTION requirements if one ECCS Subsystem is inoperable. Under the requirements of Specification 3.0.3, if both the required ECCS Subsystems are inoperable, within one hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one Spray System is inoperable. Under the requirements of Specification 3.0.3, if both the required Containment Spray Systems are inoperable, within one hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN within the subsequent 24 hours.

INSERT

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and, (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION Statements.

The intent of this provision is to i--ure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION Statements of the appropriate specifications.

(2)

It is acceptable to initiate and complete a reduction in OPERATIONAL MODES in a shorter time interval than required in the ACTION statement and to add the unused portion of this allowable out-of-service time to that provided for operation in subsequent lower OPERATION*MODE(S). Stated allowable out-of-service times are applicable regardless of the OPERATIONAL MODE(S) in which the inoperability is discovered but the times provided for achieving a mode reduction are not applicable if the inoperablity is discovered in a mode lower than the applicable mode. For example if the Containment Spray System was discovered to be inoperable while in STARTUP, the ACTION Statement would allow up to 156 hours to achieve COLD SHUTDOWN. If HOT STANDBY is attained in 16 hours rather than the allowed 78 hours, 140 hours would still be available before the plant would be required to be in COLD SHUTDOWN. However, if this system was discovered to be inoperable while in HOT STANDBY, the 6 hours provided to achieve HOT STANDBY would not be additive to the time available to achieve COLD SHUTDOWN so that the total allowable time is reduced from 156 hours to 150 hours. Insert 3

3.0.5 This specification delineated the applicability of each specification to Unit 1 and Unit 2 operation.

APPLICABILITY

BASES

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for SPECIAL TEST EXCEPTIONS need only be performed when the SPECIAL TEST EXCEPTION is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Conditions for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Conditions for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

Items may be determined inoperable during use, during surveillance tests, or in acordance with this specification. Therefore, 'ACTION' statements are entered when the Surveillance Requirements should have been performed rather than at the time it is discovered that the tests were not performed.

(3)

APPLICABILITY

BASES

4.0.5 This specification ensures 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 will 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. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

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 these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

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. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device 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.

INSERT

4.0.6 This specification delineates the applicability of the surveillance activities to Unit 1 and Unit 2 operations.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most retrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.6% Δ ~~delta~~ k/k is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% ~~delta~~ Δ k/k shutdown margin provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the FSAR accident and transient analysis.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -3.9×10^{-4} Δ ~~delta~~ k/k/°F. The MTC value of -3.0×10^{-4} Δ ~~delta~~ k/k/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -3.9×10^{-4} Δ ~~delta~~ k/k/°F.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of each fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, ~~(3) the P-12 interlock is above its setpoint,~~ 3(~~4~~) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (~~5~~) the reactor pressure vessel is above its minimum RT_{NDT} temperature. 4

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% ~~delta~~ k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 5106 gallons of 20,000 ppm borated water from the boric acid storage tanks or 75,000 gallons of 2000 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one boron injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% ~~delta~~ k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,000 ppm borated water from the boric acid storage tanks or 9690 gallons of 2000 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

2

REACTIVITY CONTROL SYSTEMS

BASES

BORATION SYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all centrifugal charging pumps except the required OPERABLE pump to be inoperable below 323°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of ~~this~~ section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) ~~limit~~ the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg} greater than or equal to 541°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

Group Demand position can be determined from 1) the group step counters; 2) the plant computer, or 3) for control rods, the P to A Converter at the rod control cabinet

3

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core evaluation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

(3)

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the ±5% target band about the target flux difference, during rapid plant THERMAL POWER changes, control rod motion will cause the AFD to deviate outside of the target band. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached subsequently (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for 2 or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

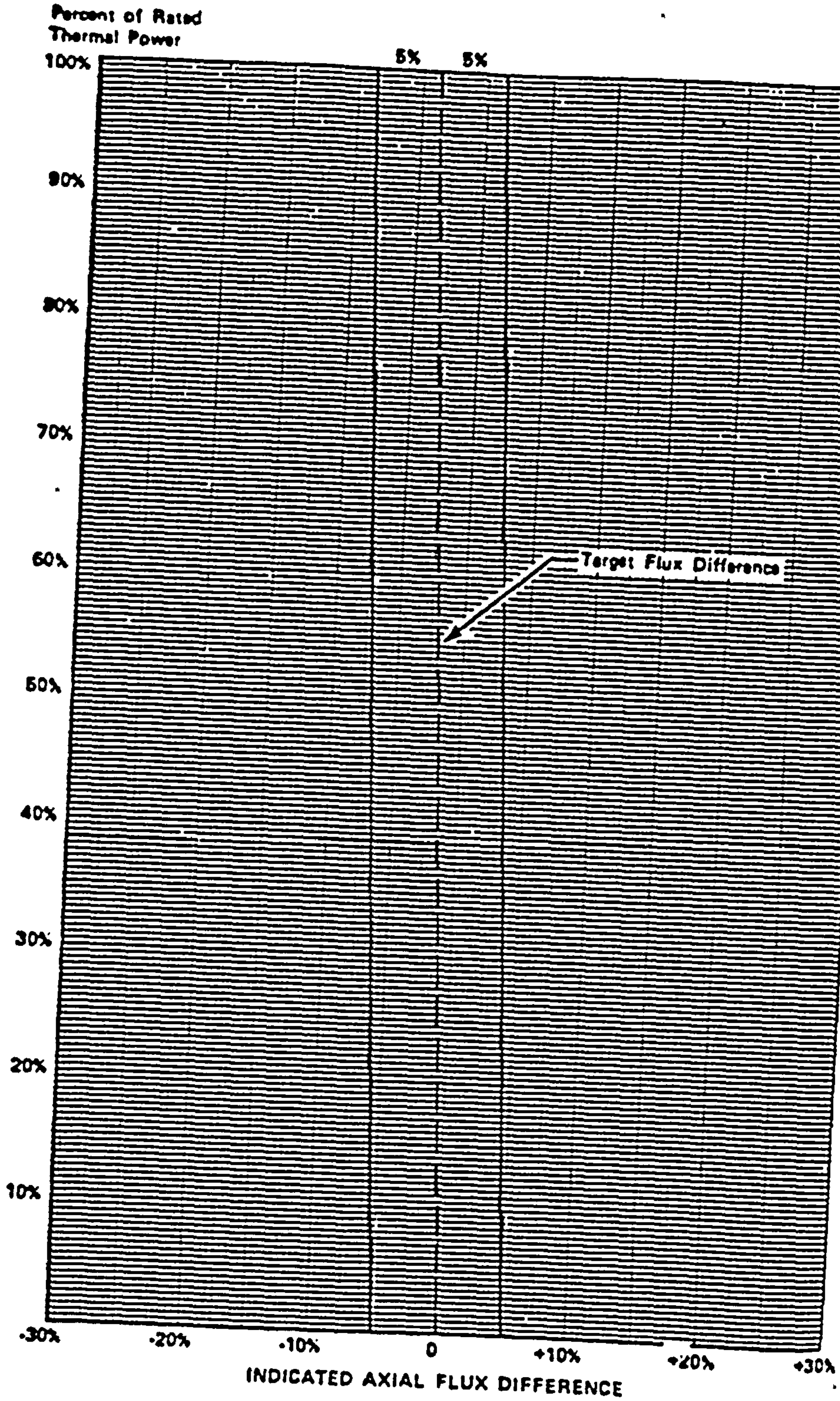

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

BASES

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

1. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position,

2. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6,

3. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained, and

4. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions 1. through 4. above are maintained. As noted on Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R, as calculated per Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is the value used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed. ~~R_2, as defined, allows for the inclusion of a penalty for Rod Bow on DNBR only~~. Thus, knowing the "as measured" values of $F^N_{\Delta H}$ and RCS flow allows

POWER DISTRIBUTION LIMITS

BASES

HEAT FLUX HOT CHANNEL FACTOR, etc. (Continued)

for "trade offs" in excess of R equal to 1.0 for the purpose of offsetting the Rod Bow DNBR penalty.

(2)

Insert→ ~~Fuel rod bowing reduces the value of DNB ratio. Sufficient credit is available to offset this reduction. This credit comes from generic design margins totaling 9.1% and 3% margin in the difference between the 1.3 DNBR safety limit and the minimum DNBR calculated for the Complete Loss of Flow event. The penalties applied to $F^N_{\Delta H}$ to account for Rod Bow (Figure 3.2-4) as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979 and W-8691 Rev. 1 (partial rod bow test data)~~.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3. Measurement errors of 3.5% for RCS total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

The Radial Peaking Factor $F_{xy}(Z)$, is measured periodically to provide additional assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F^{RTP}_{xy}) as provided in the (3) Radial Peaking Factor Limit Report per Specification 6.9.1.~~1~~8 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margin totaling 9.1% DNBR is derived from the difference between the design and required values on the following items: (a) design DNBR limit, (b) grid spacing multiplier, (c) thermal diffusion coefficient, (d) DNBR spacer factor multiplier and (e) pitch reduction. The rod bow penalty is calculated with the method described in WCAP-8691, Revision 1, and is completely compensated by the available margin of 9.1%.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. ~~A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted~~. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The ~~two~~ 2- hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 ~~percent~~ % for each percent of tilt in excess of 1%.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

BASES

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and Engineered Safety Features Actuation System instrumentation and interlocks ensure that: (1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The Engineered Safety Feature Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those engineered safety feature components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Feature Actuation System to mitigate the consequences of a steam line break or loss of coolant accident: (1) safety injection pumps start and automatic valves position, (2) reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) turbine trip, (9) auxiliary feedwater pumps start and automatic valve position, (10) containment cooling fans start and automatic valves position, (11) component cooling water pumps start and automatic valves position, ~~and (12) control room isolation and ventilation systems start~~.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates turbine trip, closes main feedwater valves on T_{avg} below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows safety injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of safety injection.

REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

P-11 On increasing pressurizer pressure, P-11 automatically reinstates safety injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of safety injection actuation on low pressurizer pressure.

P-12 On increasing ~~primary~~ reactor coolant loop temperature, P-12 automatically reinstates safety injection actuation on high steam flow coincident with either low-low T_{avg} or low steam line pressure, and provides an arming signal to the steam dump system. On decreasing ~~primary~~ reactor coolant loop temperature, P-12 allows the manual block of safety injection actuation on high steam flow coincident with either low-low T_{avg} or low steam line pressure and automatically removes the arming signal from the steam dump system.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the ~~reactor~~ core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

INSTRUMENTATION

BASES

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR ~~50~~ Part 100. The instrumentation is consistent with the recommendations of ~~Safety~~ Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1977

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteri~~a~~on 19 of 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. The normal plant instrument channels specified are suitable for use as post-accident instruments. This capability is consistent with NUREG-~~0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations.~~" 0737, "Clarification of TMI Action Plan," November 1980.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983, and

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

Since the fire detectors installed in the plant are nonseismic, an inspection will be performed following a seismic event to detect any fires.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the ~~procedures~~ methodology and parameters in the ODCP to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the ~~procedures~~ methodology and parameters in the ODCP to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and ~~controlling)~~) the concentrations of potentially explosive gas mixtures in the ~~waste gas holdup~~ GASEOUS RADWASTE system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2. shall be such that concentrations as low as 1×10^{-5} µCi/cc are measurable.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

(3)

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can by prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5, with reactor coolant loops not filled, a single RHR train provides sufficient heat removal capabilit~~ies~~ y for removing decay heat; but single failure considerations and the unavailability of the steam generator as a heat removing component require that at least two RHR trains be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump with one or more RCS cold legs less than or equal to 323°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either: (1) restricting the water volume in the pressurizer and thereby providing a volume for the ~~primary~~ reactor coolant to expand into, or (2) by restricting staring of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at 110% of the valve's Set point. The relief capacity of a single safety valve is adequate to relieve

SAFETY VALVES (Continued)

any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System (relief valves) provides a diverse means of protection against RCS overpressurization at low temperature

During operation, all pressurizer ~~C~~ode safety valves must be OPERABLE to prevent the RCS from being pressurized above its ~~S~~afety ~~L~~imit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no ~~R~~eactor trip until the first Reactor Trip System ~~T~~rip ~~S~~etpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control ~~R~~eactor ~~C~~oolant ~~S~~ystem pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power-operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer ~~C~~ode safety valves. Each PORV has remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

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

REACTOR COOLANT SYSTEM

BASES

STEAM GENERATORS (Continued)

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.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the ~~primary~~ reactor coolant system and the secondary coolant system (~~primary~~ reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a ~~primary~~ reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a wastage defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be ~~promptly~~ reported to the Commission as a pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

special report –

3

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are functionally consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

~~The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.~~

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the ~~accident~~ safety analyses.

The total steam generator tube leakage limit ~~of~~ 1 gpm for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 ~~limits~~ dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

INSERT

OPERATIONAL LEAKAGE (Continued)

The 1 gpm leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

Insert

REACTOR COOLANT SYSTEM

BASES

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

10 CFR dose guideline values

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the *Site Boundary* will not exceed an appropriately small fraction of Part 100 ~~limits~~ following a steam generator tube rupture accident in conjunction with an assumed steady state ~~primary~~-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent ~~interim~~ limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Diablo Canyon site, such as *Site Boundary* location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1 microCurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the *Site Boundary* by a factor of up to 20 following a postulated steam generator tube rupture.

The reporting of any cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1 microCurie/gram DOSE EQUIVALENT

SPECIFIC ACTIVITY (Continued)

I-131 will allow sufficient time for Commission evaluation of the circumstances before reaching the 800 hour limit. INSERT

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

1). The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the ~~first full-power~~ service period specified thereon:

a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation, and

b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2). These limit lines shall be calculated periodically using methods provided below,

3). The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4). The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 560°F, and

(3)

The sample analysis for determining the gross specific activity and Ē can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several ←Insert→ reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month. Alternatively, gamma spectroscopy may be used.

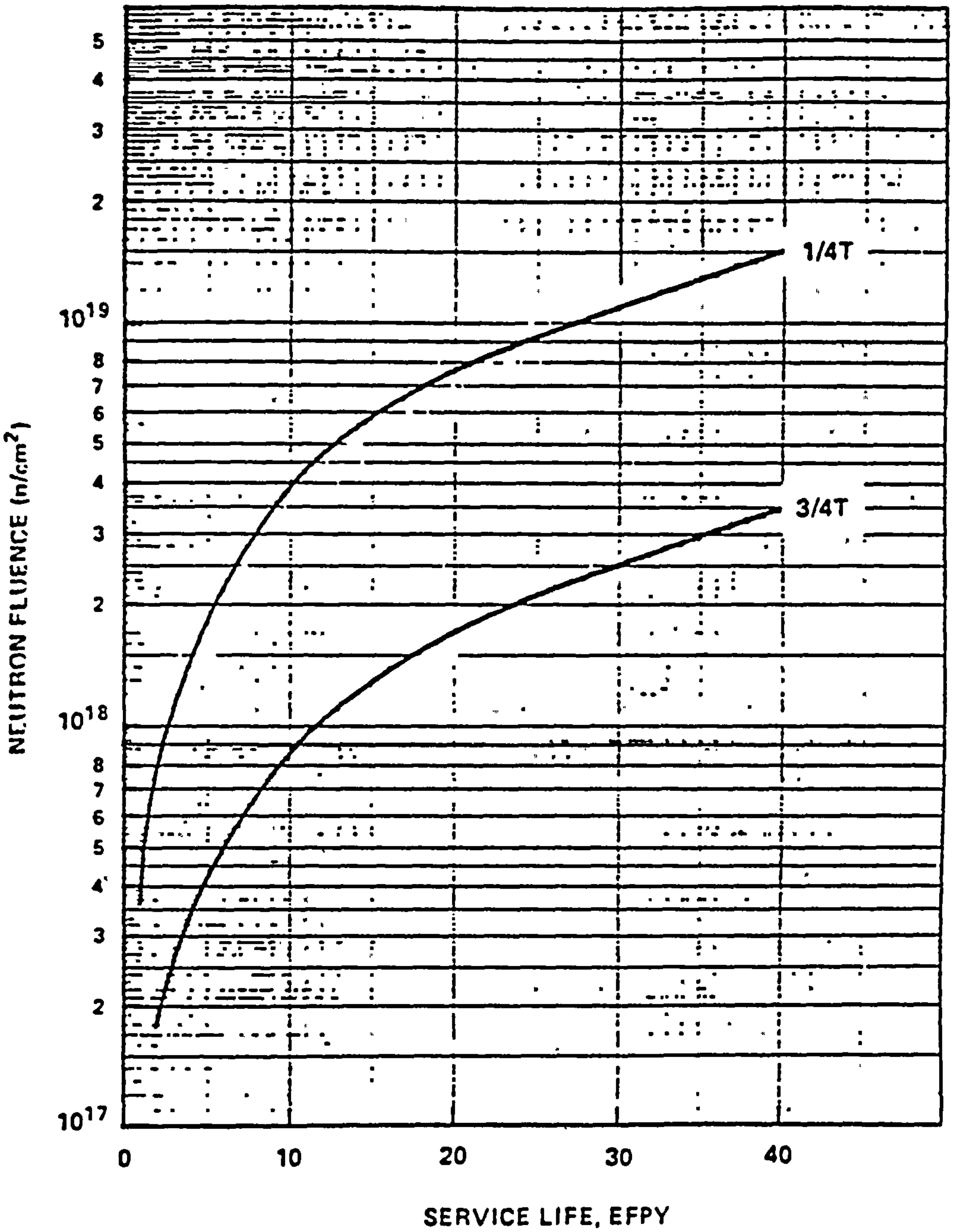

FIGURE B 3/4 4-1

FAST NEUTRON FLUENCE (E>1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

TABLE B 3/4 4-1a

REACTOR VESSEL TOUGHNESS DATA – UNIT 1

COMPONENT	PLATE NO.	MATERIAL TYPE	Cu (WT%)	P (WT%)	NDTT °F	MINIMUM 50 FT-LB/35 MIL TEMP °F LONG	MINIMUM 50 FT-LB/35 MIL TEMP °F TRANS	RT_{NDT} °F	AVERAGE UPPER SHELF FT-LB LONG	AVERAGE UPPER SHELF FT-LB TRANS
CL.HD.DOME	B4108	A533B1			-30	51	71*	11		72*
CL.HD.SEG.	B4109-1	A533B1			0	53	73*	13		81*
CL.HD.SEG.	B4109-2	A533B1			-10	50	70*	10		89*
CL.HD.SEG.	B4109-3	A533B1			-20	50	70*	10		79*
CL.HD.FLG.	B4102	A508,2			53*	30	50*	53		103*
VES.SH.FLG.	B4101	A508,2			35*	-5	15*	35		99*
INLET NOZ.	B4103-1	A508,2			60*	17	37*	60		77*
INLET NOZ.	B4103-2	A508,2			60*	27	47*	60		75*
INLET NOZ.	B4103-3	A508,2			43*	10	30*	43		108*
INLET NOZ.	B4103-4	A508,2			48*	2	22*	48		106*
OUTLET NOZ.	B4104-1	A508,2			60*	-13	7*	60		77*
OUTLET NOZ.	B4104-2	A508,2			43*	- 3	17*	43		74*
OUTLET NOZ.	B4104-3	A508,2			54*	-12	8*	54		86*
OUTLET NOZ.	B4104-4	A508,2			60*	30	50*	60		84*
UPPER SHL.	B4105-1	A533B1	0.12	0.010	10	68	88*	28		80*
UPPER SHL.	B4105-2	A533B1	0.12	0.008	0	49	69*	9		74*
UPPER SHL.	B4105-3	A533B1	0.14	0.010	0	54	74*	14		81*
INTER.SHL.	B4106-1	A533B1	0.14	0.013	-10	57	40	-10	134	116
INTER.SHL.	B4106-2	A533B1	0.13	0.013	-10	36	57	- 3	132	114
INTER.SHL.	B4106-3	A533B1	0.10	0.011	10	70	90*	30	119	77.5*
LOWER SHL.	B4107-1	A533B1	0.13	0.011	-10	59	75	15	127	110
LOWER SHL.	B4107-2	A533B1	0.12	0.010	-10	64	80	20	127	108
LOWER SHL.	B4107-3	A533B1	0.12	0.010	-50	52	38	-22	135	116
BOT.HD.SEG.	B4111-1	A533B1			-20	33	53*	- 7		82*
BOT.HD.SEG.	B4111-2	A533B1			-40	16	36*	-14		90*
BOT.HD.SEG.	B4111-3	A533B1			-40	21	41*	-19		85*
BOT.HD.SEG.	B4110	A533B1			-10	60	80*	20		75*

*Estimated per NRC Standard Review Plan Section 5.3.2.

INSERT

TABLE B 3/4.4-1b

REACTOR VESSEL TOUGHNESS DATA-UNIT 2

COMPONENT	PLATE NO.	MATERIAL TYPE	Cu (WT%)	P (WT%)	NDTT °F	MINIMUM 50 FT-LB/35 MIL TEMP °F LONG	MINIMUM 50 FT-LB/35 MIL TEMP °F TRANS	RT_{NDT} °F	AVERAGE UPPER SHELF FT-LB LONG	AVERAGE UPPER SHELF FT-LB TRANS
Cl.Hd.Dome	B5457	A533BCL1			-20	35	55*	-5	135	89*
Cl.Hd.Seg.	B5456-1	A533BCL1			-50	23	43*	-17	134	87*
Cl.Hd.Seg.	B5456-2	A533BCL1			-20	62	82*	22	131	83*
Cl.Hd.Seg.	B5456-3	A533BCL1			-20	15	35*	-20	124	81*
Cl.Hd.Flg.	B5452	A508CL2			20	15	35*	20	151	98*
Ves.Sh.Flg.	B5451	A508CL2			-10	-10	10*	-10	158	103*
Inlet Noz.	B5461-1	A508CL2			-20	23	43*	-17	116	75
Inlet Noz.	B5461-2	A508CL2			-20	- 2	18*	-20	119	77*
Inlet Noz.	B5461-3	A508CL2			-40	-45	-25*	-40	127	83*
Inlet Noz.	B5461-4	A508CL2			-40	-48	-28*	-40	129	84*
Outlet Noz.	B5462-1	A508CL2			-50	- 4	16*	-44	145	94
Outlet Noz.	B5462-4	A508CL2			-40	-10	10*	-40	137.5	89*
Outlet Noz.	B5462-2	A508CL2			-40	14	34*	-26	135.5	88*
Outlet Noz.	B5462-3	A508CL2			-50	17	37*	-23	131.5	85*
Upper Shl.	B5453-1	A533BCL1			0	85	88	28	92	82
Upper Shl.	B5453-3	A533BCL1			-10	45	65*	5	136.5	89*
Upper Shl.	B5011-1	A533BCL1			10	40	60*	0	110	72*
Inter.Shl.	B5454-1	A533BCL1	0.15	0.010	-40	14	112	52	128	91
Inter.Shl.	B5454-2	A533BCL1	0.14	0.012	0	60	127	67	113	99
Inter.Shl.	B5454-3	A533BCL1	0.15	0.012	-40	30	93	33	129	90
Lower Shl.	B5455-1	A533BCL1	0.14	0.010	-20	42	45	-15	134	112
Lower Shl.	B5455-2	A533BCL1	0.14	0.011	0	25	45	0	137	122
Lower Shl.	B5455-3	A533BCL1	0.10	0.010	0	55	75	15	128	100
Bot.Hd.Seg.	B5009-2	A533BCL1			-10	110	130*	70	85	55*
Bot.Hd.Seg.	B5009-3	A533BCL1			-20	-12	8*	-20	131	84
Bot.Hd.Seg.	B5009-1	A533BCL1			0	88	108*	48	95	62*
Bot.Hd.Seg.	B5010	A533BCL1			-30	20	40*	-20	114	74

*Estimated per NRC Standard Review Plan Section 5.3.2.

TABLE B 3/4 4-1b

IDENTIFICATION OF REACTOR VESSEL BELTLINE REGION BASE MATERIAL

Component	Plate No.	Heat No.	Material Spec. No.	Composition (Wt. %) C	Mn	P	S	Si	Ni	Mo	Cu
Inter. Shell	B4106-1	C2884	A533BC1.1	.25	1.34	.013	.015	.21	.53	.45	.14
Inter. Shell	B4106-2	C2854	A533BC1.1	.18	1.32	.013	.015	.23	.50	.46	.13
Inter. Shell	B4106-3	C2793	A533BC1.1	.20	1.33	.011	.012	.25	.46	.46	.10
Lower Shell	B4107-1	C3121	A533BC1.1	.25	1.36	.011	.014	.24	.56	.48	.13
Lower Shell	B4107-2	C3131	A533BC1.1	.24	1.32	.010	.013	.23	.56	.46	.12
Lower Shell	B4107-3	C3131	A533BC1.1	.19	1.38	.010	.013	.26	.56	.46	.12

TABLE B 3/4 4-1c

FRACTURE TOUGHNESS PROPERTIES OF UNIT NO. 1 REACTOR VESSEL BELTLINE REGION BASE MATERIAL

				Maximum Inner Wall End-of-Life				
					ΔRT_{NDT} (°F)		ΔUse (FT - LB)	
Plate No.	T_{NDT} °F	RT_{NDT} °F	Average use Ft - LB	Fluence N/Cm²	R.G. 1.99	W	R.G. 1.99	W**
B4106-1	-10	-10	116	2.6×10^{19}	202	130	33	28
B4106-2	-10	- 3	114	2.6×10^{19}	185	125	31	26
B4106-3	10	30*	77.5*	2.6×10^{19}	121	98	18.5	20
B4107-1	-10	15	110	2.6×10^{19}	169	125	30	26
B4107-2	-10	20	103	2.6×10^{19}	145	115	27	24
B4107-3	-50	-22	116	2.6×10^{19}	145	115	30	24

* Estimated from data in the longitudinal direction per NRC Standard Review Plan Section 5.3.2.
** Estimated per WCAP-8291, dated May 1974

TABLE B 3/4 4-1d

IDENTIFICATION OF REACTOR VESSEL BELTLINE REGION WELD METAL

Weld Location	Weld Process	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	Composition (Wt. %)* C	Mn	P	S	Si	Mo	Ni	Cr	Cu
Nozzle Shell to Inter. Shell Circle Seam 8-442	Sub-Arc	B-4 Mod	13253	Linde 1092	3774	.15	1.83	.013	.015	.06	.45	.72	.04	.07
Inter. Shell Long Seams 2-442 A, B&C	Sub-Arc	B-4 Mod.	27204	Linde 1092	3724	.21	1.81	.014	.010	.08	.54	1.07	.07	.12
Inter. Shell to Lower Shell Circle Seam 9-442	Sub-Arc	B-4 Mod.	21935	Linde 1092	3869	.20	2.00	.012	.008	.05	.54	.71	-	-
Lower Shell Long Seams 3-442 A, B&C	Sub-Arc	B-4 Mod.	27204	Linde 1092	3774	.21	1.81	.014 .018**	.010	.08	.54	1.07	.07	.12 .35**

*Analysis performed on bare wire, not on as deposited weld metal.

**Estimated maximum P and Cu composition in deposited metal of weld 3-442C. This is the limiting metal, and is the material basis for the heatup and cooldown curves.

3

INSERT

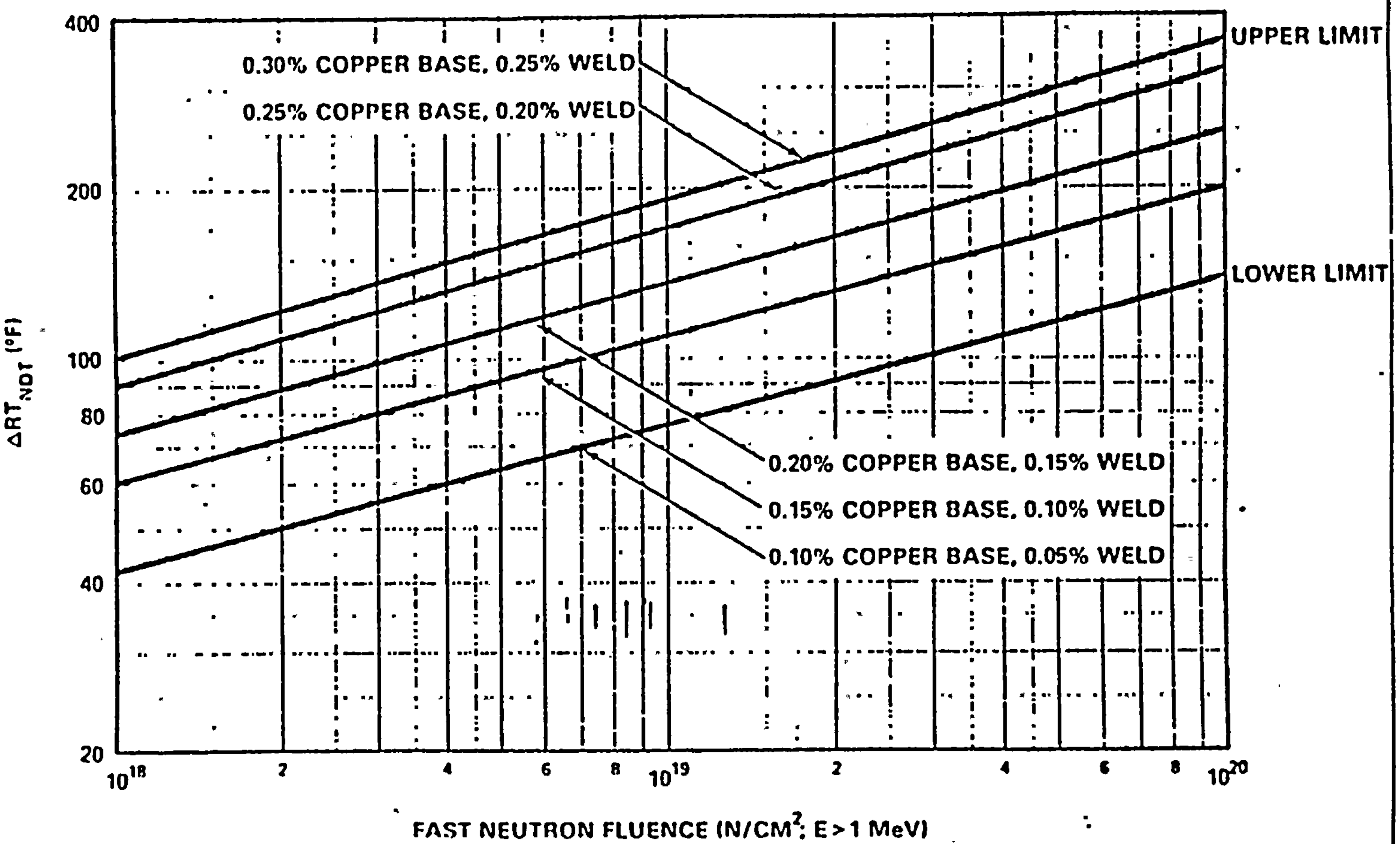

FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSELS EXPOSED TO 550°F

TABLE B 3/4 4-2

FRACTURE TOUGHNESS PROPERTIES OF REACTOR VESSEL BELTLINE REGION WELD METAL

Seam No.	T_{NDT}* °F	Energy at 10°F Ft-LBs	RT_{NDT}* °F	Average USE Ft-LBs	Fluence N/CM2	Maximum Inner Wall End-of-Life ΔRT_{NDT} (°F)** R.G. 1.99	W	ΔUSE (Ft-LBs) R.G. 1.99	W
8-442	0	74,63,82	0	-	1.8×10^{18}	140	157	(Not determined since upper shelf energy and copper contents are not available)	
2-442 A&B	0	71,78,60	0	-	1.4×10^{19}	300	281		
2-442 C	0	71,78,60	0	-	8.4×10^{18}	275	242		
9-442	0	62,59,60	0	-	2.6×10^{19}	340	335		
3-442 A&B	0	49,48,36	0	-	1.1×10^{19}	290	262		
3-442 C	0	49,48,36	0	-	2.6×10^{19}	340	335		

*Estimated per NRC Standard Review Plan Section 5.3.2.
**Estimated using upper limit curve in Reg. Guide 1.99 and Westinghouse trend curves.

PRESSURE/TEMPERATURE LIMITS (Continued)

5. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

~~The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."~~ insert

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of ~~4.5~~ 6 effective full power years of service life. The ~~4.5~~ 6 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1a for Unit 1 and Table B 3/4.4-1b for Unit 2. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, ~~and~~ copper ~~content~~ and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and ~~the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."~~ insert The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of ~~4.5~~ 6 EFPY.

Values of delta RT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The heatup and cooldown curves must be recalculated with the ~~delta~~ Δ RT_{NDT} determined from the surveillance capsule exceeds the calculated ~~delta~~ Δ RT_{NDT} for the equivalent capsule radiation exposure.

The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel.

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Insert

The fracture toughness testing of the ferritic materials in the reactor vessel were performed in accordance with the 1968 Edition for Unit 2 of the ASME Boiler and Pressure Vessel Code, Section III. These properties are then evaluated in accordance with the NRC Standard Review Plan.

1966 Edition for Unit 1 or

Insert

The largest value of ΔRT_{NDT} computed by either the Regulatory Guide 1.99 ~~Trend Curves from the Regulatory Guide 1.99~~, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials" or the Westinghouse Copper Trend Curves shown by Figure B 3/4.4-2.

PRESSURE/TEMPERATURE LIMITS (Continued)

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in ~~WCAP-7924-A.~~ the following paragraphs

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nilductility reference temperature, RT_{NDT}, is used and this includes the radiation induced shift, ~~delta~~ Δ RT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp[0.0145(T-RT_{NDT} + 160)] \quad (1)$$

Where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

PRESSURE/TEMPERATURE LIMITS (Continued)

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature of the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value. The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

PRESSURE/TEMPERATURE LIMITS (Continued)

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

PRESSURE TEMPERATURE LIMITS (Continued)

~~Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.~~

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operation limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements. (2)

INSERT I

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for the ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.11 REACTOR VESSEL HEAD VENTS

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

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

The function, capabilities, and testing requirements of the Reactor Coolant System Vent Systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

(3)

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 323°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization for all anticipated transients.

Insert

The Maximum Allowed PORV Setpoint for the LTOPs will be modified, if required, based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the ~~reactor~~ core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

The requirement to maintain the RHR Suction Valves 8701 and 8702 in the locked closed condition in MODES 1, 2 and 3 provides assurance that a fire could not cause inadvertent opening of these valves when the RCS is pressurized to near operating pressure. These valves are not part of an ECCS subsystem.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all centrifugal charging pumps and safety injection pumps except the required OPERABLE charging pump to be inoperable below 323°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the Boron Injection System as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS ~~system~~ cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 21,000 ppm boron.

(3)

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 ~~PRIMARY~~ CONTAINMENT

(3)

3/4.6.1.1 CONTAINMENT INTEGRITY

~~Primary~~ CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the ~~limits~~ dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, Pa. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 47 psig during LOCA conditions.

(2)

¶ The maximum peak pressure expected to be obtained from a LOCA event is 46.~~65~~ psig. This includes the limit of 0.3 psig for initial positive containment pressure. ~~will limit~~ the total pressure ~~to 46.65 psig which~~ is less than design pressure and is consistent with the ~~accident~~ analyses.

3/4.6.2 2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration, ensure that: (1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and (2) corrosion effects on components within containment are minimized. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6 2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment fan cooler units ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post LOCA conditions, and (3) adequate mixing of the containment atmosphere following a LOCA to prevent localized accumulations of hydrogen from exceeding the flammable limit.

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and 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 expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These Hydrogen Control Systems are functionally consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a ~~LOCA", March 1971~~. Loss-of-Coolant Accident," Revision 2, November 1978.

PLANT SYSTEMS

BASES

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each electric motor driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 440 gpm at a pressure of 1135 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 880 gpm at a pressure of 1135 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 350°F in the event of a total loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on secondary coolant system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 ~~limits~~ dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm ~~primary~~ [illegible] to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the ~~accident~~ safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements ~~are~~ is consistent with the assumptions used in the safety analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on average steam generator impact values taken at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 VITAL COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the vital component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 AUXILIARY SALTWATER SYSTEM

The OPERABILITY of the auxiliary saltwater system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the ~~accident conditions within acceptable limits~~.

safety analysis

3/4.7.5 CONTROL ROOM VENTILATION SYSTEM

The OPERABILITY of the Control Room Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. ANSI N510-19~~75~~80 will be used as a procedural guide for surveillance testing. (A)

3/4.7.6 AUXILIARY BUILDING SAFEGUARDS AIR FILTRATION SYSTEM

The OPERABILITY of the Auxiliary Building Safeguards Air Filtration System ensures that radioactive materials leaking from the ECCS equipment within the auxiliary building following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating to maintain low humidity for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-19~~75~~80 will be used as a procedural guide for surveillance testing. (1C)

3/4.7.7 SNUBBERS

All snubbers are required 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 are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same desgn features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Staff Review Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

SNUBBERS (Continued)

The visual inspection frequency is based upon maintaining a constant level of snubber protection during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given type and is determined by the number of inoperable snubbers found during an inspection of each type. In order to establish the inspection frequency for each type of snubber, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber of that type could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results 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.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability, one of three functional testing methods are used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

SNUBBERS (Continued)

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.) The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the Fire Suppression Systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The Fire Suppression System consists of the water system, spray and/or sprinklers, CO_2, Halon and fire water hose stations. The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility Fire Protection Program.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirement of the Fire Suppression Systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying the weight of the tanks.

FIRE SUPPRESSION SYSTEMS (Continued)

In the event the Fire Suppression Water System becomes inoperable, prompt corrective measures must be taken since this system provides the major fire suppression capability of the plant. ~~The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued operation of the nuclear plant.~~

3/4.7.10 FIRE BARRIER PENETRATIONS

The functional integrity of the fire barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The fire barrier penetrations are a passive element in the Facility Fire Protection Program and are subject to periodic inspections.

Fire barrier penetrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when a barrier is not functional, either: (1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or (2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established, until the barrier is restored to functional status.

3/4.7.11 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause loss of its OPERABILITY. The temperature limits include allowance for an instrument error of 1° F.

3/4.7.12 ULTIMATE HEAT SINK

The OPERABILITY of the Component Cooling Water (CCW) System and the components that it cools is ensured if the CCW temperature remains equal to or less than 132°F during any condition assumed in the safety analysis. One CCW heat exchanger is required in service when the ocean temperature is 64°F or less. Two CCW heat exchangers are required in service when the ocean temperature is greater than 64°F. If the reactor coolant temperature is less than 350°F (MODE 4), one CCW heat exchanger in service is adequate even if the ocean temperature is greater than 64°F.

3/4.7.13. FLOOD PROTECTION

The breakwaters (east and west) provide flood protection to the safety-related auxiliary salt water (ASW) pumps located in the intake structure. The ASW pumps would be flood protected for the events up to and including the probable maximum tsunami and maximum credible wave, if the breakwaters are OPERABLE at a level of 0 feet mean lower low water (MLLW) or above. However, substantial degradation of the breakwaters below MLLW level may result in the ASW pump being flooded under the design basis flood and would require corrective actions to be taken. The ongoing surveillance of the breakwaters will ensure that proper flood protection is provided the ASW pumps.

3/4.8 ELECTRICAL POWER SYSTEM

BASES

3/4.8.1, 3/4.8.2 ~~AND~~ 3/4.8.3 A.C. SOURCES, D.C. SOURCES ~~AND~~ ONSITE POWER DISTRIBUTION ~~SYSTEMS~~

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for: (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR Part 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources is consistent with the initial condition assumptions of the safety analyses and is based upon maintaining sufficient redundancy of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of one onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generators as a source of emergency power, are also OPERABLE, and that at least two auxiliary feedwater pumps are OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term "verify" as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the facility status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, where applicable, and 1.137 "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979, where applicable. The Third Diesel Generator Unit is designed to respond to a Safety Injection Signal from Either Unit 1 or Unit 2. If the capability to respond to a Safety Injection Signal from one Unit is maintained during surveillance testing on the other Unit, the third (common) Diesel Generator Unit shall be considered to be OPERABLE for that Unit.

ELECTRIC POWER SYSTEMS

BASES

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION ~~SYSTEMS~~ (Continued)

The Surveillance Requirements for demonstrating the OPERABILITY of the batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-3 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-3 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

The OPERABILITY of the motor-operated valves thermal overload protection and bypass devices ensures that these devices will not prevent safety related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturers brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

(3)

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the ~~accident~~ safety analysis.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that: (1) manipulator cranes will be used for movement of control rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and (3) the core internals and ~~pressure~~ reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - ~~SPENT FUEL STORAGE~~ FUEL HANDLING BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control assembly and associated handling tool, except the movable fuel handling building walls, over other fuel assemblies in the spent fuel pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of the fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses. The movable fuel handling building walls travel on rollers over the spent fuel pool and have been designed to remain in place during postulated seismic events.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) train be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor ~~pressure~~ vessel below 140°F as required during the REFUELING MODE and (2) sufficient coolant circulation is maintained ~~thru~~ through the reactor core to mimimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR trains OPERABLE when there is less than 23 feet of water above the reactor ~~pressure~~ vessel flange ensures that a single failure of the operating RHR train will not result in a complete loss of residual heat removal capability. With the reactor vessel heat removed and 23 feet of water above the reactor ~~pressure~~ vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR train, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment ventilation penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and SPENT FUEL POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL HANDLING BUILDING VENTILATION SYSTEM

The limitations on the Fuel Handling Building Ventilation System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. Transfer of system operation into the iodine removal mode (exhaust through HEPA filters and charcoal absorbers) is initiated automatically by either the new fuel storage or spent fuel pool area radiation monitors required by Specification 3.3.3. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.9.13 SPENT FUEL SHIPPING CASK MOVEMENT

The restriction on spent fuel shipping cask movement ensure that the contents of no more than ~~twenty~~ fuel assemblies with at least 1000-hour decay time will be ruptured in the event of a spent fuel shipping cask accident. The dose consequences of this accident are within the dose guideline values of 10 CFR Part 100. ~~The location of Racks 5 and 6 is~~ shown in Figure 9.1-3 of the FSAR. Sfe:

~~During the natural circulation tests, the condensate storage tank water level may be reduced to less than 178,000 gallons. In order to provide a backup water supply during the tests, the fire water tank and its flow path to the auxiliary feedwater pump shall be determined OPERABLE prior to and during the test.~~

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to: (1) measure control rod worth and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the reactor core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is, at times, necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the Moderator Temperature Coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 which may in turn, cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

~~3/4.10.4 REACTOR COOLANT LOOPS~~

~~This special test exception permits reactor criticality under no flow~~ conditions ~~and is required in order to perform certain startup and PHYSICS TESTS~~ while at low THERMAL POWER levels ~~and permits natural circulation tests~~ both at low thermal ~~power and during cooldown.~~

3/4.10.~~5~~4 POSITION INDICATION SYSTEMS-SHUTDOWN

This special test exception permits the ~~P~~position ~~I~~indication ~~S~~systems to be inoperable during rod drop time measurement. This exception is required since the data necessary to determine the rod drop time is derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small, compared to the normal voltage and therefore can not be observed if the ~~P~~position ~~I~~indication ~~S~~systems remain OPERABLE.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

to UNRESTRICTED AREAS

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents ~~from the site~~ will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water ~~outside the site~~ UNRESTRICTED AREA will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, to ~~an individual~~ a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

⇐ ADD PARAGRAPHS 1 & 2

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." The dose calculations in the ODCP implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of ~~an individual~~ a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCP for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification SLCT applies to the release of radioactive materials in liquid effluents from each ~~reactor~~ unit at the site. ~~For units with~~ shared Radwaste Treatment Systems, the ~~liquid effluents from the shared system are proportioned among the units sharing that system.~~ SLCT wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

Insert Paragraphs (1) & (2)
p. B 3/4 11-1

This specification applies to the release of radioactive materials in liquid effluents from all units at the site.

▶-insert-

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), ~~Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Annal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).~~

Currie, L.A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the

C

Insert

~~This specification applies to the release of radioactive materials in lic effluents from each unit at the site. When shared Radwaste Treatment Systems used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.~~

Insert Paragraph p. B 3/4 11-2

This specification applies to the release of radioactive materials in liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

3/4.11.1.3 LIQUID RAD WASTE TREATMENT SYSTEM

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

Add Paragraph

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes or walls capable of holding the tank contents [illegible] Treatment Sys

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply in an unrestricted area.

[illegible]

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 ~~for~~ unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a member of the public ~~an individual~~ in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For members of the public ~~individuals~~ who may at times be within the site boundary, the occupancy ~~of the individual~~ will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a member of the public ~~an individual~~ at or beyond the site boundary to less than or equal to 500 mrem/year to the ~~total~~ body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to ~~an~~ a child ~~infant~~ via the ~~cow-milk-infant~~ inhalation pathway to less than or equal to 1500 mrem/year. ~~for the nearest cow to the plant.~~

DOSE RATE (Continued)

radioactive materials in

This specification applies to the release of gaseous effluents from all units ~~reactors~~ at the site. ~~For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.~~

ADD 1st PARAGRAPH

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC ~~an individual~~ through appropriate pathways is unlikely to be substantially underestimated. The dose calculations methodology and parameters established in the ODCP for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCP equations provided for determining the air doses at the site boundary are based upon the historical average atmospheric conditions.

ADD 2nd PARAGRAPH

Iodine-131, Iodine-133, Tritium, and

3/4.11.2.3 DOSE - ~~RADIOIODINES~~, RADIOACTIVE MATERIALS IN PARTICULATE FORM ~~AND RADIONUCLIDES OTHER THAN NOBLE GASES~~

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCP calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC ~~an individual~~ through appropriate pathways is unlikely to be substantially underestimated. The ODCP calculational methodology and parameters ~~methods~~ for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual

(1)

Paragraph (1)
Insert p. B 3/4 11-3

The required detection capabilities for radioactive material in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), ~~Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination-Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975)~~.

Currie, L.A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the

Paragraph (2)

~~This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.~~

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

RADIOACTIVE EFFLUENTS

BASES

~~DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES (Continued)~~

Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications ~~for radioiodines, radioactive materials in particulate form and radionuclides other than noble gases~~ are dependent ~~on~~ the existing radionuclide pathways to man, ~~in the unrestricted area~~. The pathways ~~which~~ were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

ADD 1st PARAGRAPH →

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the GASEOUS RADWASTE SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

ADD 2ND PARAGRAPH →

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the ~~waste gas holdup system~~ is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

Insert Paragraph 1 p. B 3/4 11-4

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared Radwaste Treatment Systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

Insert

~~This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatmer Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases canno accurately be ascribed to a specific unit. An estimate should be made of t contributions from each unit based on input conditions, e.g., flow rates an radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producin units sharing the Radwaste Treatment System. For determining conformance t LCOs, these allocations from shared Radwaste Treatment Systems are to be ad to the releases specifically attributed to each unit to obtain the total releases per unit.~~

Insert Paragraph 2 p. B 3/4 11-4

Insert

~~This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatmen Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases canno accurately be ascribed to a specific unit. An estimate should be made of t contributions from each unit based on input conditions, e.g., flow rates an radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producin units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be ad to the releases specifically attributed to each unit to obtain the total releases per unit.~~

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared Radwaste Treatment Systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

RADIOACTIVE EFFLUENTS

Replace with Insert A

BASES

3/4.11.2.6 GAS STORAGE TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity that provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed [illegible]

~~Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting~~ whole ~~total~~ body exposure to a MEMBER OF THE PUBLIC ~~an individual~~ at the nearest ~~exclusion area~~ boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan ~~15.7.1, "Waste Gas System Failure"~~. 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTE

~~The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped offsite.~~ This specification implements the requirement of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid, solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, or mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190 [illegible] that have been incorporated into 10 CFR Part 20 by [illegible]. The specification requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel cycle sources [illegible] ~~from plant radioactive effluents exceed twice the design objective doses of Appendix I~~. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units and from outside storage tanks are [illegible] ~~the reporting requirement level~~. The Special Report will describe a course of action which should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC ~~for 12 consecutive months~~ to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km ~~5 miles~~ must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and [illegible], is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation ~~which~~ that is part of the nuclear fuel cycle.

Insert Ⓐ p. B 3/4 11-5

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

implements Section IV.B.2. of Appendix I to 10 CFR Part 50 and

3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this specification provides repr: measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides, ~~which~~ that lead to the highest potential radiation exposures of ~~individuals~~ MEMBERS OF THE PUBLIC resulting from the ~~station~~ plant operation. This monitoring program thereby supplements the Radiological Effluent Monitoring Program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. Guidance for this program is given by the Radiological Assessment Branch Technical Position, Environmental, Revision 1, November 1979. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

ADD TWO PARAGRAPHS.

~~The detection capabilities required by Table 4.12-1 are state-of-the-art for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report. Since there is no sediment on the shoreline in the vicinity of the plant, no sediment sampling is required.~~

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY ~~unrestricted areas~~ are identified and that modifications to the ~~monitoring~~ Radiological Environmental Monitoring Program are made if required by the results of this census. The best ~~survey~~ information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m² ~~square feet~~ provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were ~~used~~ made: 1) ~~that~~ 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/~~square meter~~ m².

①

Insert Ⓐ p. B 3/4 11-5

Insert

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in Currie, L.A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually), ~~Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975)~~.

RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are ~~reasonably~~ valid for the purposes of Section IV.E.2 of Appendix I to 10 CFR Part 50.

SECTION 5.0

DESIGN FEATURES

3

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be shown in Figure 5.1-3.

The definition of UNRESTRICTED AREA used in implementing these Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a. Insert

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

~~RADIOACTIVE DISCHARGE POINTS~~

~~5.1.3 The radioactive discharge points shall be as shown in Figure 5.1-3.~~

INSERT

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 140 feet.

b. Nominal inside height = 212 feet.

c. Minimum thickness of concrete walls = 3.6 feet.

d. Minimum thickness of concrete roof = 2.5 feet.

e. Minimum thickness of concrete floor pad = 14.5 feet.

f. Nominal thickness of steel liner, wall and dome = 3/8 inch.

g. Nominal thickness of steel liner, base = 1/4 inch.

h. Net free volume = 2.55×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 ~~The reactor~~ Containment ~~building~~ is designed and shall be maintained for a maximum internal pressure of 47 psig and a temperature of 271°F, coincident with a Double Design Earthquake.

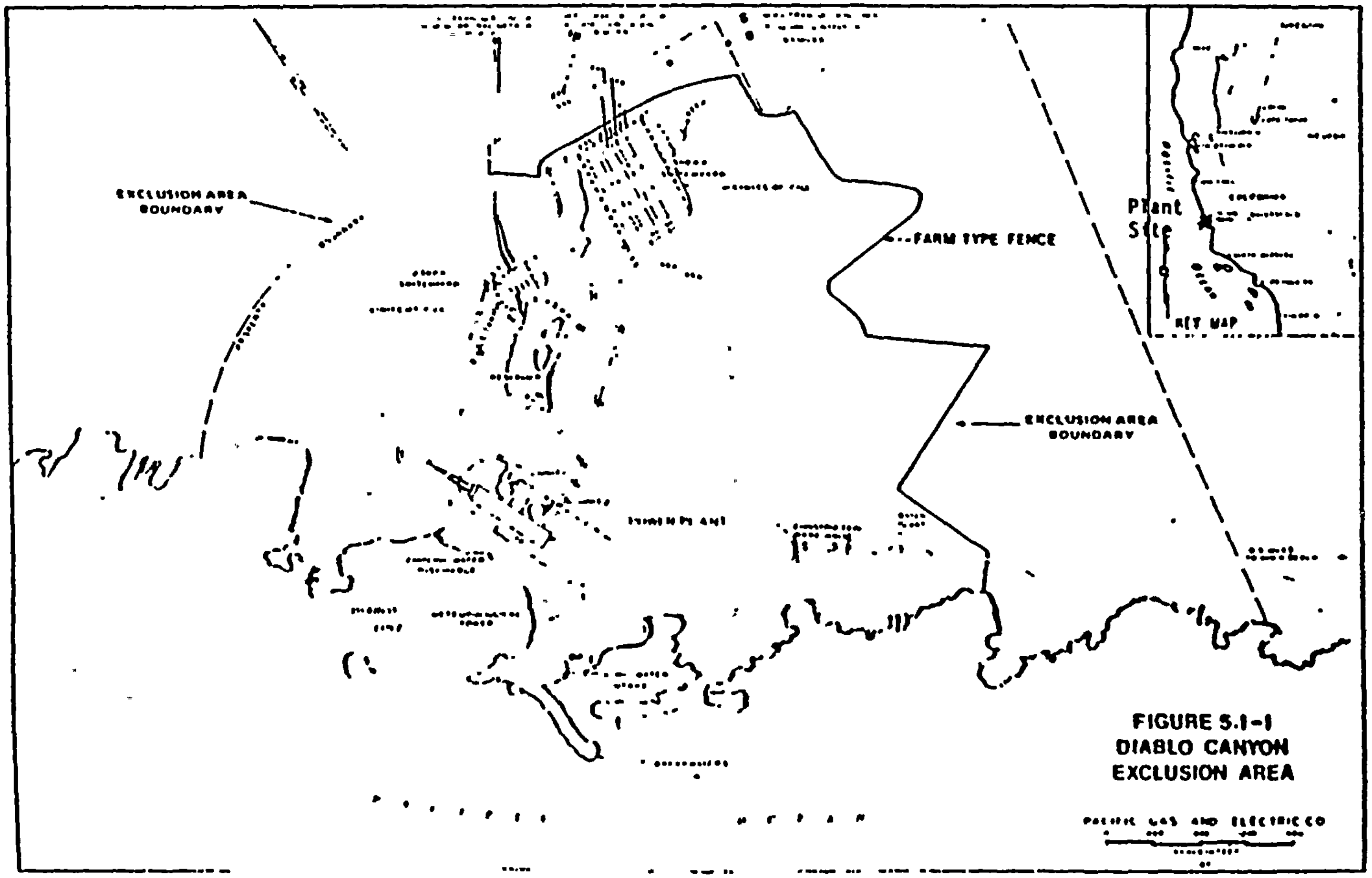

FIGURE 5.1-1
DIABLO CANYON
EXCLUSION AREA

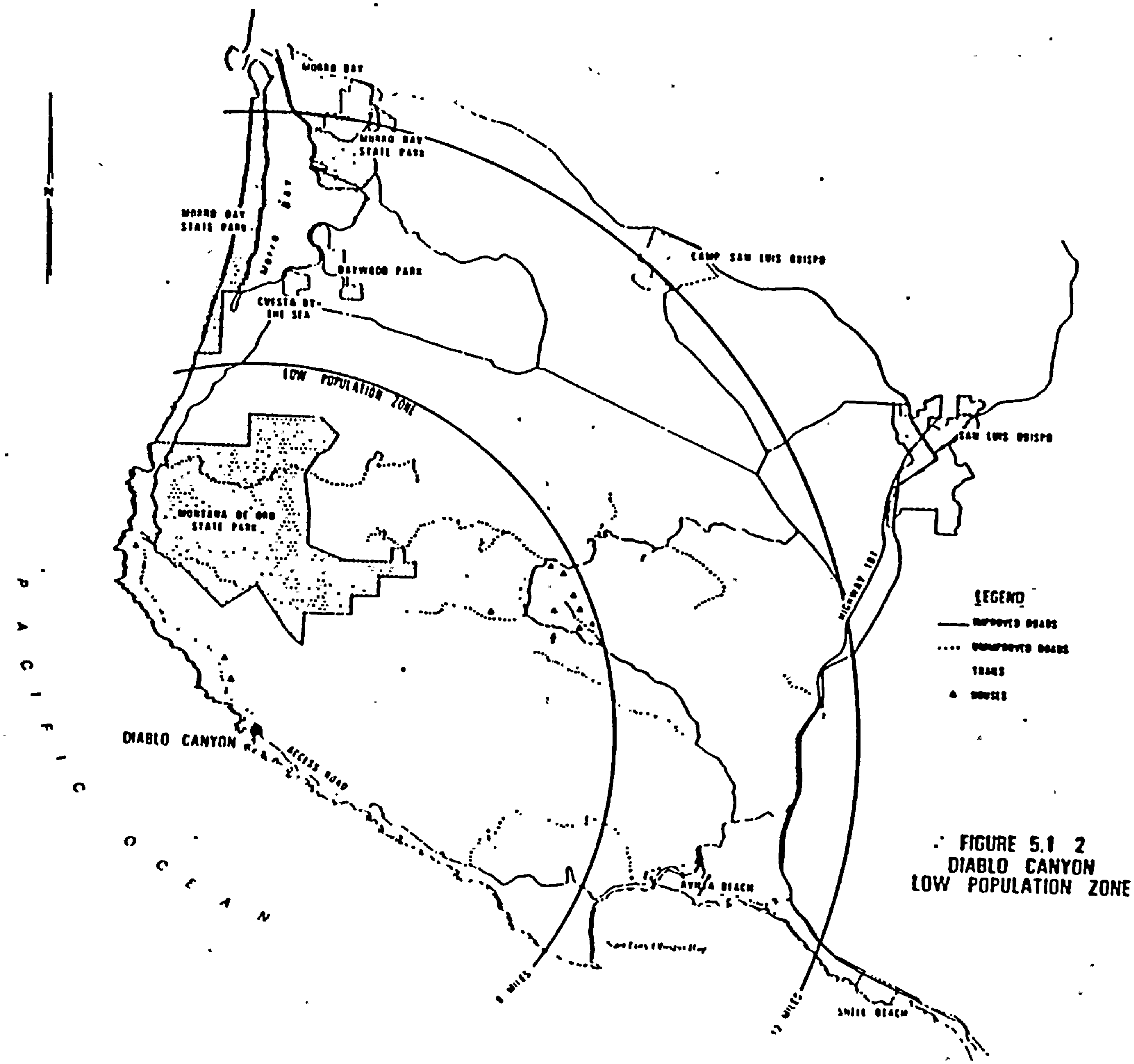

FIGURE 5.1 2
DIABLO CANYON
LOW POPULATION ZONE

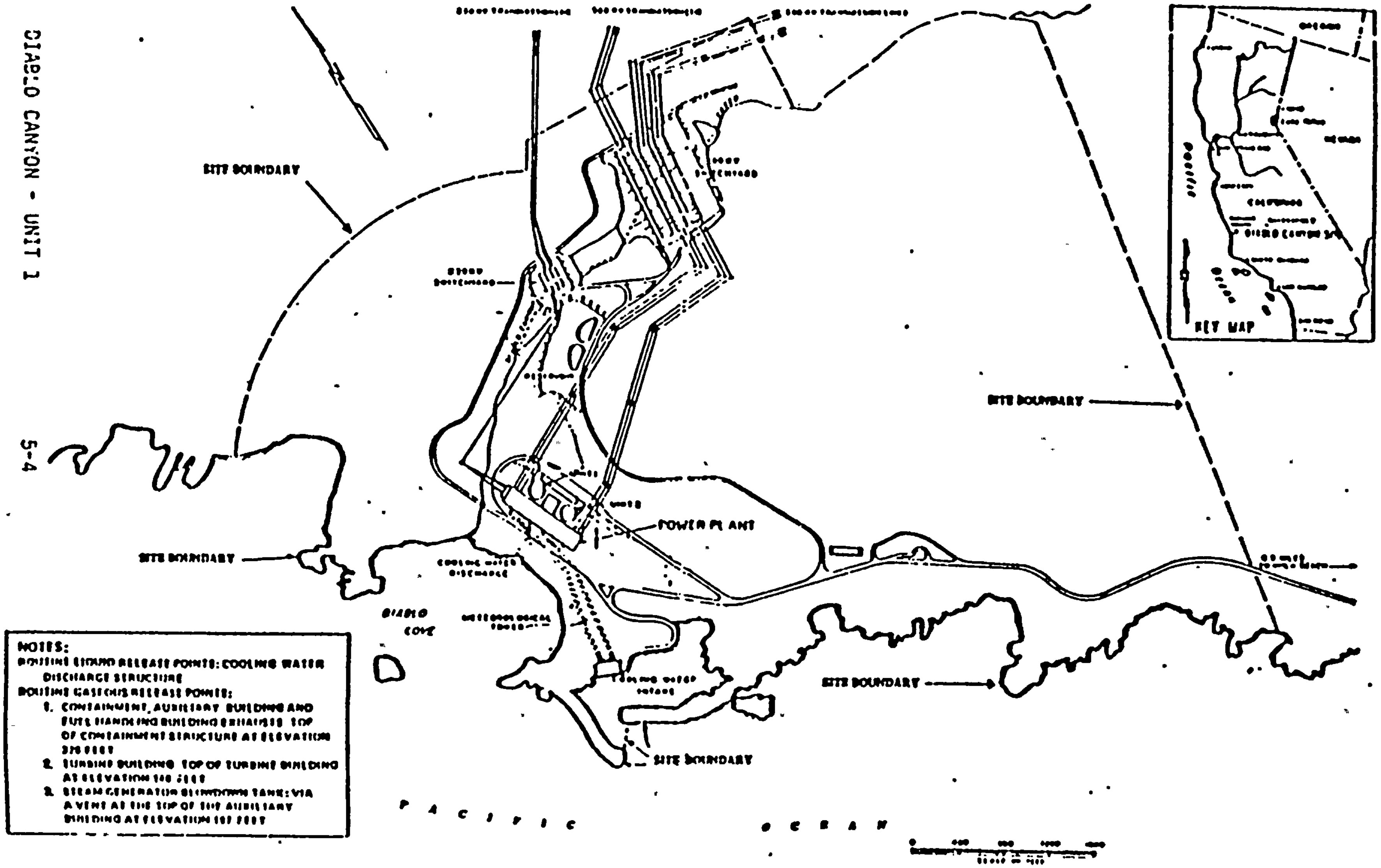

FIGURE 5.1-3

RADIOACTIVE DISCHARGE POINTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with (Zircaloy -4). Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 17~~1~~6 grams uranium. The initial core loading shall have a maximum enrichment of 3.15 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full-length and no part length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

3

5.4.2 The total water and steam volume of the reactor coolant system is ~~13,164~~ 12,811 ± 100 cubic feet at a nominal T_{avg} of 576°F for Unit 1 and 12,903 ± 100 cubic feet at a nominal T_{AVG} of 577°F for Unit 2.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.6% ~~delta~~ k/k for uncertainties as described in Section 9.1 of the FSAR. ~~and~~

b. A nominal 21 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.90 when flooded with unborated water.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 133.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 270 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	250 Heatup and Cooldown Cycles	200°F to 550°F to 200°F except for pressurizer, which is 200°F to 650°F to 200°F
	100 Loss of load cycles, with turbine trip and without immediate reactor trip	Above 15% RTP to 0% RTP
	50 cycles of loss of all offsite electrical power	Loss of all offsite power with turbine trip and initial power level at 100% RTP
	100 cycles of loss of flow in one reactor coolant loop	Loss of one reactor coolant pump with initial power level at 100% RTP
	500 reactor trip cycles	100% to 0% RTP
	12 inadvertent auxiliary spray actuation cycles	Spray water temperature differential > 320°F and < 560°F
	60 leak tests	Pressurized to > 2500 psig coincident with no pressurization of the secondary side of steam generators
	10 hydrostatic pressure tests	Pressurized to ≥ 3107 psig
Secondary Coolant System	10 hydrostatic pressure tests each steam generator	Pressurized to > 1356 psig coincident with the primary side at 0 psig.
	10 Turbine roll tests	Turbine roll on RCP heat resulting in plant cooldown rate > 100°F/hr.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall ~~unit~~ plant operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room Command function. A management directive to this effect signed by the Vice-President, Nuclear Power Generation shall be reissued to all ~~station~~ plant personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The Offsite Organization for ~~unit~~ plant management and technical support shall be as shown on Figure 6.2-1.

PLANT
~~UNIT~~ STAFF

6.2.2 The ~~Unit~~ Plant Organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Operator shall be in the Control Room;

c. A Health Physics Technician* shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site Fire Brigade of at least five members* shall be maintained onsite at all times. The Fire Brigade shall not include the Shift Supervisor and the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

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*The Health Physics Technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

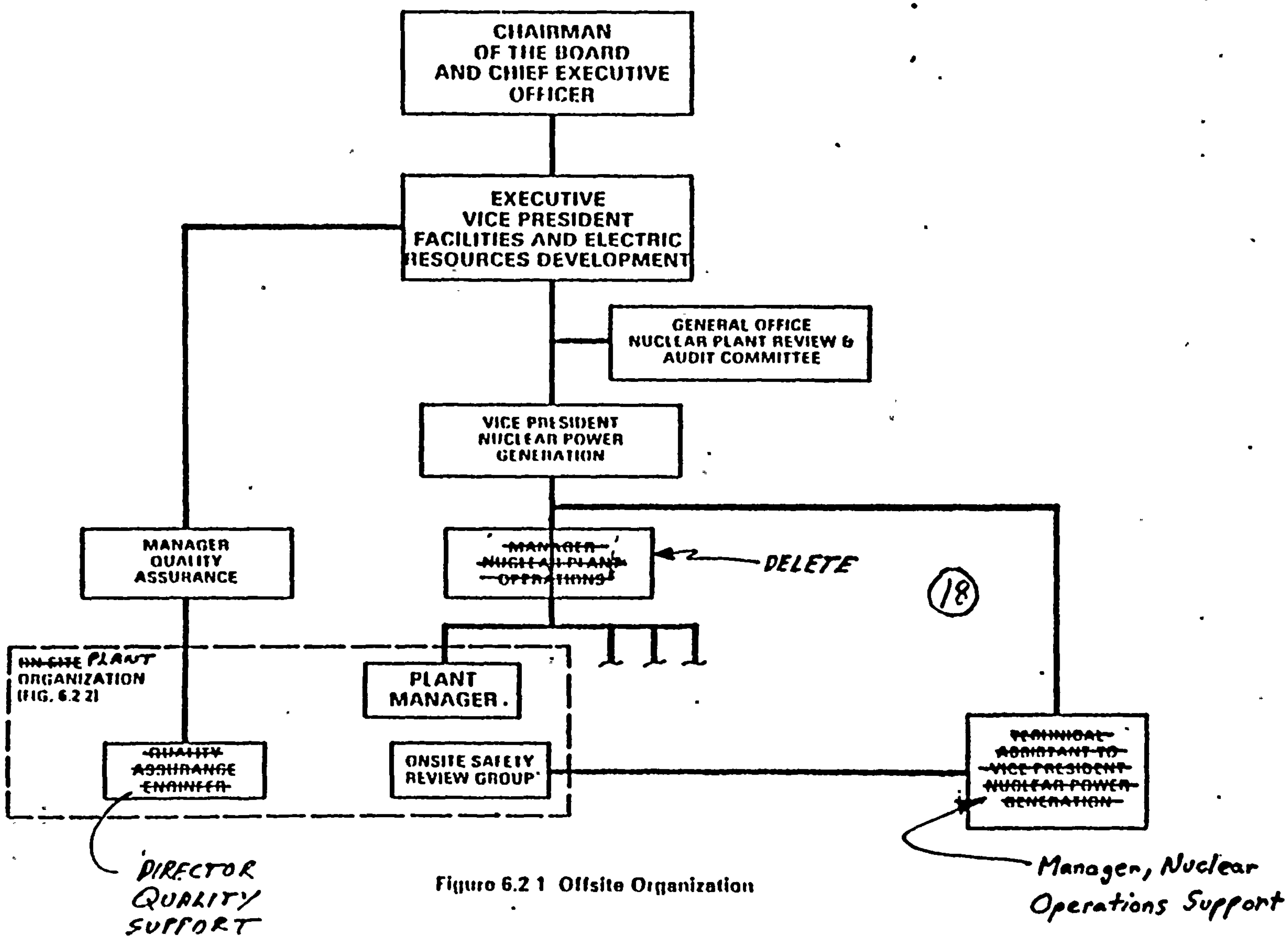

Figure 6.2 1 Offsite Organization

(19)

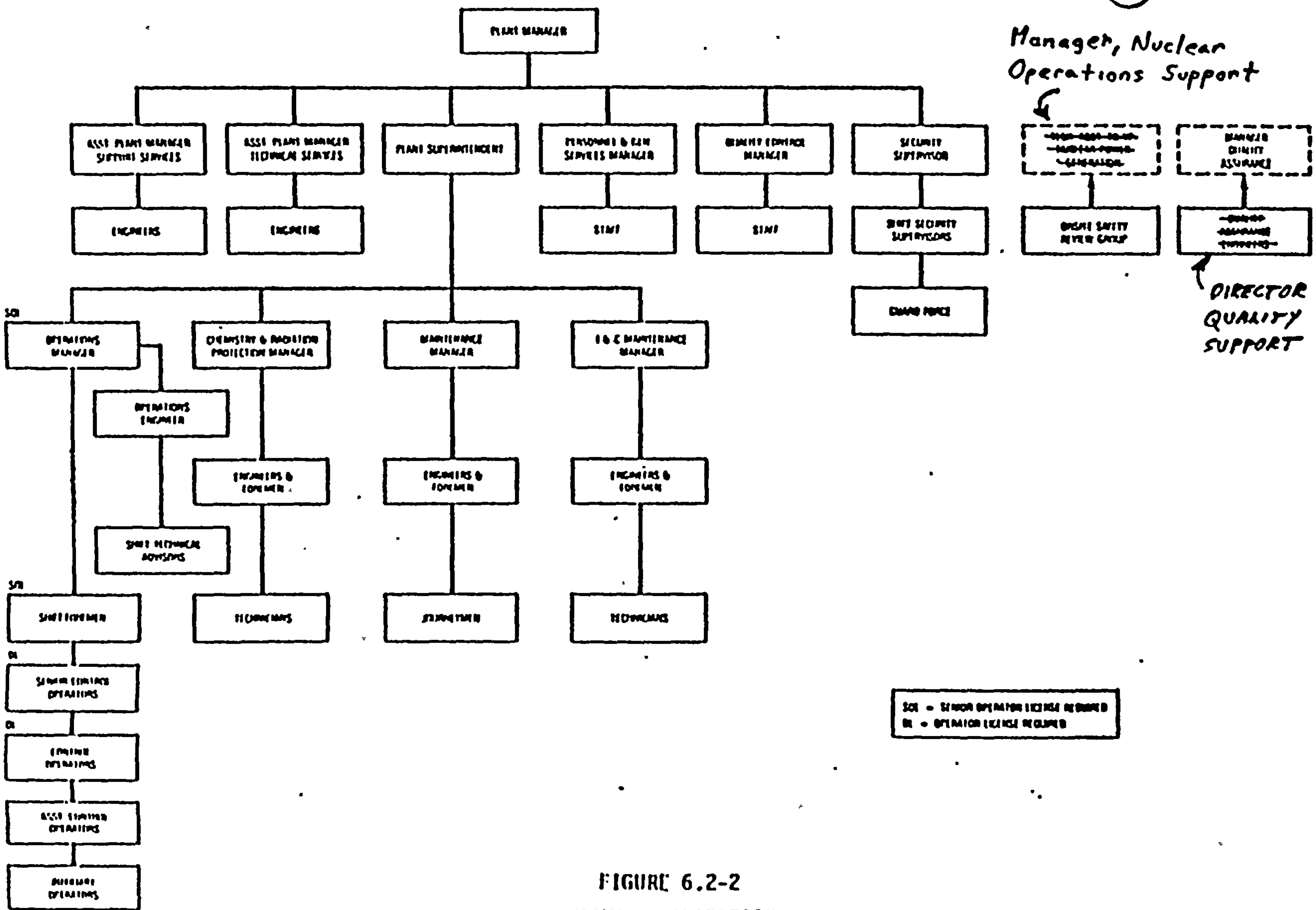

FIGURE 6.2-2

~~UNIT~~ ORGANIZATION
PLANT

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1	1
SOL	1	None
OL	2	1
AO	2	1
STA	1*	None

INSERT

SS - Shift Supervisor with a Senior Operators License ~~on Unit 1~~
SOL - Individual with a Senior Operator License ~~on Unit 1~~
OL - Individual with an Operator License ~~on Unit 1~~
AO - Auxiliary Operator
STA - Shift Technical Advisor

The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 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. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license shall be designated to assume the control room command function.

INSERT

*The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, OR 4	BOTH UNITS IN MODE 5 OR 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3 OR 4 AND ONE UNIT IN MODE 5 OR 6 OR DEFUELED
SS	1	1	1
SOL	1	none[b]	1
OL	3[a]	2[a]	3[a]
AO	3[a]	3[a]	3[a]
STA	1*	none	1*

←Insert

SS - Shift Supervisor with a Senior Operators License
SOL - Individual with a Senior Operator License
OL - Individual with an Operator License
AO - Auxiliary Operator
STA - Shift Technical Advisor

a/ At least one of the required individuals must be assigned to the designated position for each unit.

←Insert b/ At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during Core Alterations on either unit, who has no other concurrent responsibilities.

PLANT
~~UNIT~~ STAFF (Continued)

plant

f. Administrative procedures shall be developed and implemented to limit the working hours of ~~unit~~ staff who perform safety-related functions; e.g., licensed Senior Operators, licensed Operators, Health Physicists, auxiliary operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time;

3) A break of at least 8 hours should be allowed between work periods, including shift turnover time; and

4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Superintendent or his designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Superintendent or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

6.2.3 ONSITE SAFETY REVIEW GROUP (OSRG)

FUNCTION

plant REPORTABLE EVENTS

6.2.3.1 The OSRG shall function to examine ~~unit~~ operating characteristics, NRC issuances, industry advisories, ~~Licensee Event Reports~~ and other sources of plant design and operating experience information, including plants of similar design which may indicate areas for improving plant safety.

INSERT From Page 6-6

COMPOSITION

6.2.3.2 The OSRG shall be composed of at least five engineers located on site.

RESPONSIBILITIES

6.2.3.3 The OSRG shall be responsible for maintaining surveillance of ~~unit~~ Plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

(3) ~~AUTHORITY~~

To Page 6-5 ~~6.2.3.4~~ The OSRG shall make plant detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving ~~unit~~ safety to the ~~Technical Assistant to the Vice President, Nuclear Power Generation.~~ ~~Director, Nuclear Administration and Support Services.~~ Manager, Nuclear Operations Support (18)

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4 ~~.1~~ The Shift Technical Advisor shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the unit.

6.3 ~~UNIT~~ PLANT STAFF QUALIFICATIONS

6.3 ~~.1~~ Each member of the ~~unit~~ plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions ~~and~~ the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, except for the Chemistry and Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, for a Radiation Protection Manager. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the

6.4 TRAINING

6.4.1 A retraining and replacement training program for the ~~unit~~ plant staff shall be maintained under the direction of a designated member of the facility staff and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience ~~identified by the OSRG~~. (3)

6.5 REVIEW AND AUDIT

6.5.1 PLANT STAFF REVIEW COMMITTEE (PSRC)

FUNCTION

6.5.1.1 The Plant Staff Review Committee shall function to advise the Plant Manager on all matters related to nuclear safety.

*Not responsible for sign-off function.

COMPOSITION

6.5.1.2 The Plant Staff Review Committee shall be composed of the:

Chairman:	Plant Manager
Member:	Plant Superintendent
Member:	Operations Manager
Member:	Assistant Plant Manager, Support Services
Member:	Quality Control Manager
Member:	Maintenance Manager
Member:	Assistant Plant Manager, Technical Services
Member:	Chemistry and Radiation Protection Manager

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PSRC Chairman to serve on a temporary basis; however, no more than one alternate shall participate as a voting member in PSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PSRC shall meet at least once per calendar month and as convened by the PSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum ~~A~~ quorum of the PSRC necessary for the performance of the PSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and two members one of which may be an alternate.

RESPONSIBILITIES

6.5.1.6 The Plant Staff Review Committee shall be responsible for:

a. Review of: (1) all procedures required by Specification 6.8 and changes thereto, (2) all programs required by Specification 6.8 and changes thereto, (3) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix "A" Technical Specifications;

d. Review of all proposed changes or modifications to Plant ~~unit~~ systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the ~~Manager~~ Vice, Pres. Nuclear Power Generation ~~of Nuclear Plant Operations~~ and to the Chairman of the General Office Nuclear Plant Review and Audit Committee;

f. Review of ~~events requiring 24-hour written notification to the Commission;~~ all REPORTABLE EVENTS;

RESPONSIBILITIES (Continued)

g. Review of ~~unit~~ plant operations to detect potential nuclear safety hazards;

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the General Office Nuclear Plant Review and Audit Committee;

i. Review of the Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the General Office Nuclear Plant Review and Audit Committee or the Plant Manager, as appropriate;

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the General Office Nuclear Plant Review and Audit Committee or the Plant Manager, as appropriate;

k. Review of ~~every~~ any accidental unplanned, or uncontrolled ~~onsite release of~~ radioactive release ~~material to the environs~~ including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the ~~Manager of the Nuclear Plant Operations~~ Vice President, Nuclear Power Generation and to ~~the~~ GONPRAC; and

3

l. Review of changes to the PROCESS CONTROL PROGRAM, ~~OFFSITE DOSE CALCULATION PROCEDURE~~ ODCP, ERMP, and Radwaste Treatment Systems.

~~AUTHORITY~~

6.5.1.7 The Plant Staff Review Committee shall:

a. Recommend to the Plant Manager written approval or disapproval of items considered under Specification 6.5.1.6a. through d. above;

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. through e. above constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the ~~Manager of Nuclear Plant Operations~~ Vice President, Nuclear Power Generation and the General Office Nuclear Plant Review and Audit Committee of disagreement between the PSRC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The Plant Staff Review Committee shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the ~~Manager of Nuclear Plant Operations~~ Vice President, Nuclear Power Generation and ~~Chairman of~~ the General Office Nuclear Plant Review and Audit Committee.

6.5.2 GENERAL OFFICE NUCLEAR PLANT REVIEW AND AUDIT COMMITTEE (GONPRAC)

FUNCTION

6.5.2.1 The General Office Nuclear Plant Review and Audit Committee shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

COMPOSITION

2 INSERT FROM PAGE 6-12

6.5.2.2 ~~The~~ GONPRAC shall be composed of the following:

Chairman:	Vice President, Nuclear Power Generation
Vice Chairman:	~~Manager, Nuclear Plant Operations*~~ Assistant to the Vice President, Nuclear Power Generation
Member:	Project Manager, Diablo Canyon*
Member:	Manager, Quality Assurance
Member:	Technical Assistant to Vice President, Nuclear Power Generation
Member:	Chief Mechanical and Nuclear Engineer
Member:	Manager, Station Construction
Member:	Manager, Nuclear Operations Support

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the GONPRAC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in GONPRAC activities at any one time.

~~*Members for these positions shall have an academic degree in an engineering or physical science field and a minimum of five years technical experience, of which three years shall be in their respective field of expertise.~~

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CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the GONPRAC Chairman to provide expert advice to ~~the~~ GONPRAC.

MEETING FREQUENCY (3)

6.5.2.5 ~~The~~ GONPRAC shall meet at least once per calendar quarter during the initial year of ~~unit~~ operation following fuel loading and at least once per 6 months thereafter. plant

QUORUM (3)

6.5.2.6 A quorum of ~~the~~ GONPRAC necessary for the performance of the GONPRAC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four GONPRAC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the ~~unit~~ plant.

REVIEW

6.5.2.7 The GONPRAC shall review:

a. The safety evaluations for: (1) changes to procedures, equipment or systems and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of ~~unit~~ plant equipment that affect nuclear safety;

(2) g. All REPORTABLE EVENTS; ~~Events requiring 24 hour written notification to the Commission~~;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components that could affect nuclear safety; and

i. Reports and meetings minutes of the Plant Staff Review Committee and the Onsite Safety Review Group. (1)

f. The fire protection equipment and program implement at least once per 12 months utilizing either a qualif offsite licensee fire protection engineer or an outsi independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

AUDITS

6.5.2.8 Audits of ~~unit~~ plant activities shall be performed under the cognizance of ~~the~~ GONPRAC. These audits shall encompass:

a. The conformance of ~~unit~~ plant operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training and qualifications of the entire ~~unit~~ plant staff at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in plant ~~unit~~ equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The fire protection program and implementing procedures at least once per 24 months by qualified personnel;

~~f. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 21 months utilizing either qualified offsite license personnel or an outside fire protection firm~~;

~~g. An inspection and audit of the fire protection and loss prevention program shall be performed by an qualified outside fire consultant at least once per 36 months;~~

g ~~h~~. Any other area of ~~unit~~ plant operation considered appropriate by ~~the~~ GONPRAC or the Executive Vice President, Facilities and Electric Resources Development;

h ~~i~~. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

i ~~j~~. The ~~OFFSITE DOSE CALCULATION PROCEDURE~~ ODCP and ERMP and implementing procedures at least once per 24 months;

j ~~k~~. The PROCESS CONTROL PROGRAM and implementing procedures for ~~solidification~~ processing and packaging of radioactive wastes at least once per 24 months; and

k ~~l~~. The performance of activities required by the Quality Assurance Program ~~to meet the criteria of Regulatory Guide 4.15, December 1977~~ for effluent and environmental monitoring, at least once per 12 months.

(3)

~~AUTHORITY~~

Move To Page 6-9

~~6.5.2.9~~ The GONPRAC shall report to and advise the Executive Vice President, Facilities and Electric Resources Development, on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS (3)

6.5.2.~~10~~ 9 Records of GONPRAC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each GONPRAC meeting shall be prepared, approved and forwarded to the Executive Vice President, Facilities and Electric Resources Development, within 14 days following each meeting;

b. Reports of reviews encompassed by ~~Section~~ Specification 6.5.2.7 above, shall be prepared, approved and forwarded to the Executive Vice President, Facilities and Electric Resources Development, within 14 days following completion of the review; and

c. Audit reports encompassed by ~~Section~~ Specification 6.5.2.8 above, shall be forwarded to the Executive Vice President, Facilities and Electric Resources Development, and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE ~~OCCURRENCES~~ EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE ~~OCCURRENCES.~~ EVENT

a. The Commission shall be notified and~~/or~~ a report submitted pursuant to the requirements of ~~Specification 6.9~~ 10 CFR 50.73; and

b. Each REPORTABLE ~~OCCURRENCE requiring 24 hour notification to the Commission~~ EVENT shall be reviewed by the PSRC and submitted to ~~the~~ GONPRAC and the ~~Manager of Nuclear Plant Operations~~ Vice President, Nuclear Power Generation the results of this review.

6.7 SAFETY LIMIT VIOLATION

6.7.~~1~~ The following actions shall be taken in the event a Safety Limit is violated:

~~a. The unit shall be placed in at least HOT STANDBY within 1 hour;~~

a. ~~b.~~ The ~~NRC Operations Center~~ NRC shall be notified by telephone as soon as possible and in all cases within 1 hour. The ~~Manager of Nuclear Plant Operations~~ Executive Vice President Facilities Electric Resources Development and ~~the~~ GONPRAC shall be notified within 24 hours;

b. ~~c.~~ A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems or structures, and (3) corrective action taken to prevent recurrence; ~~and~~

c. ~~d.~~ The Safety Limit Violation Report shall be submitted to the Commission, ~~the~~ GONPRAC and the ~~Manager of Nuclear Plant Operations~~ Executive Vice President, Facilities and Electric Resources Development within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

the emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as state in Generic Letter No. 82-33;

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, Revision 2, February 1978,

b: ~~Refueling operations~~,

~~c. Surveillance and test activities of safety related equipment~~,

c. ~~d.~~ Security Plan implementation,

d. ~~e.~~ Emergency Plan implementation,

~~f. Fire Protection Program implementation~~,

e. ~~g.~~ PROCESS CONTROL PROGRAM implementation,

f. ~~h.~~ ~~OFFSITE DOSE CALCULATION PROCEDURE~~ ODCP and ERMP implementation, and

g. ~~i.~~ Quality Assurance Program for effluent and environmental monitoring, ~~using the guidance in Regulatory Guide 4.15, December 1977~~.

6.8.2 Each procedure of Specification 6.8.1 above, and changes thereto, shall be reviewed by the PSRC and approved by the Plant Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed by the PSRC and approved by the Plant Manager within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. ~~Primary~~ Reactor Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include portions of the Recirculation Spray System, Safety Injection System, Chemical And Volume Control System, Residual Heat Removal System, RCS Sample System, and Liquid and Gaseous Radwaste Treatment Systems. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

2) Integrated leak test requirements for each system at refueling cycle intervals or less.

PROCEDURES AND PROGRAMS (Continued)

b. In-Plant Radiation Monitoring

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.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,
2) Identification of the procedures used to measure the values of the critical variables,
3) Identification of process sampling points, including monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,
4) Procedures for the recording and management of data,
5) Procedures defining corrective actions for all off-control point chemistry conditions, and
6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1) Training of personnel, and
2) Procedures for monitoring.

e. Postaccident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1) Training of personnel,
2) Procedures for sampling and analysis, and
3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS ~~AND REPORTABLE OCCURRENCES~~

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the ~~Director~~ Regional Administrator of the Regional Office of ~~Inspection and Enforcement~~ the NRC unless otherwise noted.

STARTUP REPORTS

6.9.1.1 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.

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall 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.

6.9.1.3 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 ~~3~~ three months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual Reports covering the activities of the unit as described below during the previous calendar year shall be submitted prior to March 31 of each year. The initial report shall be submitted prior to March 31 of ~~the~~ year following initial criticality.

~~6.9.1.5~~ Reports required on an annual basis shall include:

* A single submittal may be made for a multiple unit ~~station~~ Plant. The submittal should combine those sections that are common to all units at the ~~station~~ Plant.

ANNUAL REPORTS (Continued)

~~1~~. ~~A~~ tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, ~~1/~~* e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling 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 received from external sources shall be assigned to specific major work functions.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT ~~2/~~ **

6.9.1.~~6~~ 5 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

~~6.9.1.7~~ The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous ~~E~~nvironmental ~~S~~urveillance ~~R~~eports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of ~~L~~and ~~U~~se ~~C~~ensuses required by Specification 3.12.2. ~~If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.~~

INSERT

The Annual Radiological Environmental Operating Reports shall include ~~summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975, of all radiological environmental samples taken during the report period~~. In the event that some 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 as soon as possible in a supplementary report.

* ~~1/~~ This tabulation supplements the requirements of ~~Section 20.407 of~~ 10 CFR ~~Part~~ 20.407.

** ~~2/~~ A single submittal may be made for a multiple unit ~~station~~ Plant. ~~The submittal should combine those sections that are common to all units at the station~~; ~~however~~, for units with separate ~~radwaste systems, the submittal shall~~ specify the releases of radioac~~tive material from each unit.~~ Plant

Insert

(1)

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 tables and figures in the ERMP as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979.

reason for not conducting the Radiological Environmental Monitoring Program as required by Specification 3.12.1, and discussion of all deviations from the sampling schedule of Table 3.12-1; discussion of environmental sample measurements that exceed the reporting levels of Table 3.12-2 but are not the result of plant effluents, pursuant to ACTION b. of Specification 3.12.1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable. Insert

corrective action taken if the specified program is not being performed.

ADMINISTRATIVE CONTROLS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; at least two legible maps* covering ~~a map of~~ all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; ~~and~~ the results of licensee participation in the Interlaboratory Comparison Program and the required by Specification 3.12.3; ← INSERT

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT[3] **

6.9.1.~~8~~6 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

~~6.9.1.9~~ The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured) ~~on magnetic tape~~, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.*** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to ~~individuals~~ MEMBERS OF THE PUBLIC due to their activities inside the site boundary (Figure 5.1-3 ~~and 5.1-4~~) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION PROCEDURE (ODCP).

[3] ** A single submittal may be made for a multiple unit ~~station~~ plant. The submittal should combine those sections that are common to all units at the ~~station~~ plant; however, for units with separate Radwaste Systems, the submittal shall specify the releases of radioactive material from each unit.

* One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

For solid wastes, the format for table 3 in Appendix B shall be supplemented with three additional categories; class of solid wastes (as defined by 10 CFR Part 61), type of container (e.g. LSA, Type A, Type B, Large Quantity) and SOLIDIFICATION agent or absorbent (eg; cement, urea formaldehyde).

*** In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request

MEMBER OF THE PUBLIC

(i)

calendar year

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed ~~real individual~~ MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, (including doses from primary effluent pathways and direct radiation), for the previous ~~12 consecutive months~~ calendar year to show conformance with 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

~~The Semiannual Radioactive Effluent Release Reports shall include the following information for each type of solid waste shipped offsite during the report period~~:

~~a. Container volume~~,

~~b. Total Curie quantity (specify whether determined by measurement or estimate)~~,

~~c. Principal radionuclides (specify whether determined by measurement or estimate)~~,

~~d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms)~~,

~~e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and~~

~~f. Solidification agent (e.g., cement, urea formaldehyde)~~.

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents ~~on a quarterly basis~~. made during the reporting period.

INSERT

~~The Semiannual Radioactive Effluent Release Reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) made during the reporting period~~.

MONTHLY OPERATING REPORT

(3)

6.9.1.~~10~~ 7 Routine reports of operating statistics and shutdown experience, including documentation of all challenges and failures to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the ~~NRC Regional Office~~, no later than the 15th of each month following the calendar month covered by the report. of the NRC

(1)

~~Any changes to the OFFSITE DOSE CALCULATION PROCEDURE or the ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURE shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the PSRC.~~

(1)

, the ECMP

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PCP and the ODCP, pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

Insert

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specifications 3.3.3.9 or 3.3.3.10, respectively ; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specifications 3.11.1.4 or 3.11.2.6, respectively.

②

REPORTABLE OCCURRENCES

6.9.1.11 The REPORTABLE OCCURRENCES of Specifications 6.9.1.12 and 6.9.1.13 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.12 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Regional Administrator of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a Licensee Event Report form. Information provided on the Licensee Event Report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the Reactor Trip System or other systems subject to Limiting Safety System Settings to initiate the required protective function by the time a monitored parameter reaches the Setpoint specified as the Limiting Safety System Setting in the Technical Specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a Limiting Condition for Operation is less conservative than the least conservative aspect of the Limiting Condition for Operation established in the Technical Specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady-state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the Technical Specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the Safety Analysis Report or in the Bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the Safety Analysis Report or Technical Specifications Bases; or discovery during unit life of conditions not specifically considered in the Safety Analysis Report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specification 3.11.1.1 or 3.11.2.1.

k. Exceeding the limits in Specification 3.11.1.4 or 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

THIRTY-DAY WRITTEN REPORTS

6.9.1.13 The types of events listed below shall be the subject of written reports to the Regional Administrator of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a Licensee Event Report form. Information provided on the Licensee Event Report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor Trip System or Engineered Safety Feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Conditions for Operation or plant shutdown required by a Limiting Conditions for Operation.

THIRTY DAY WRITTEN REPORTS (Continued)

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in Reactor Trip Systems or Engineered Safety Feature Systems.

d. Abnormal degradation of systems other than those specified in Specification 6.9.1.12c. above designed to contain radioactive material resulting from the fission process.

e. An unplanned offsite release of: (1) more than 1 Curie of radioactive material in liquid effluents, (2) more than 150 Curies of noble gas in gaseous effluents, or (3) more than 0.05 Curie of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

1) A description of the event and equipment involved,

2) Cause(s) for the unplanned release,

3) Actions taken to prevent recurrence, and

4) Consequences of the unplanned release.

f. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 3.12-2 when averaged over any calendar quarter sampling period.

2

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.~~14~~ 8. The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided to the NRC Regional Administrator ~~of the Regional Office,~~ with ~~a~~ copy to ~~the~~ Director, of Nuclear Reactor Regulation, Attention: Chief, ~~of the~~ Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, for all core planes containing Bank "D" control rods and all unrodded core planes at least 60 days prior to each cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it will be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission. This report is not required for the initial cycle.

SPECIAL REPORTS

6.9.2 Special Reports shall be submitted to the ~~Director~~ ~~Regional Administrator~~ STET of the ~~Office of Inspection~~ Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated:

RECORD RETENTION (Continued)

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS ~~OCCURRENCES submitted to the Commission,~~;

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications;

e. Records of changes made to procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

g. Records of training and qualification for current members of the unit staff;

h. Records of in-service inspections performed pursuant to these Technical Specifications;

i. Records of Quality Assurance activities required by the QA Manual;

RECORD RETENTION (Continued)

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the PSRC and the GONPRAC;

l. Records of analyses required by the Radiological Environmental Monitoring Program;

3

m. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.7.~~1~~ including the date at which the service life commences and associated installation and maintenance records; and

n. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of work permits for radiation (WPR). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the WPR issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Chemistry and Radiation Protection Manager in the WPR.

HIGH RADIATION AREA (Continued)

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or Health Physics supervision. Doors shall remain locked except during periods of access by personnel under an approved WPR which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the WPR, direct or remote (such as closed ciruit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

    1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

    2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

    3) Documentation of the fact that the change has been reviewed and found acceptable by the PSRC.

b. Shall become effective upon review and acceptance by the PSRC.

6.14 OFFSITE DOSE CALCULATION PROCEDURE (ODCP) and ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURE (ERMP)

6.14.1 The ODCP and ERMP shall be approved by the Commission prior to implementation. (1)

6.14.2 Licensee-initiated changes to the ODCP and/or ERMP:

a. Shall be submitted to the Commission in the ~~Monthly Operating Report~~ Semiannual Radioactive Effluent Release Report ~~within 90 days of the date~~ for the period in which the change(s) was made effective. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCP and/or ERMP to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the PSRC.

b. Shall become effective upon review and acceptance by the PSRC.

6.15 MAJOR CHANGES TO ~~RADIOACTIVE WASTE~~ LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS* ~~(Liquid, Gaseous and solid)~~ (1)

6.15.1 Licensee initiated major changes to the ~~Radioactive~~ Rad. Waste Treatment Systems (liquid, gaseous and solid):

a. Shall be reported to the Commission in the ~~Monthly Operating~~ Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PSRC. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

---

* The licensee may choose to submit the information called for in this Specification as part of the annual FSAR update.

(1)

MEMBER OF THE PUBLIC

MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Continued)

4) An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5) An evaluation of the change which shows the expected maximum exposures to ~~individual~~ in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the PSRC.

b. Shall become effective upon review and acceptance by the PSRC.

ENCLOSURE 1

PGandE COMMENTS ON NRC FINAL DRAFT COMBINED UNITS 1 AND 2 FULL POWER TECHNICAL SPECIFICATIONS

A. TECHNICAL COMMENTS

During preparation of the final draft Combined Units 1 and 2 full power Technical Specifications (T.S.), PGandE discussed with the NRC Staff the following items which were not included in the draft Combined Technical Specifications submitted to PGandE on July 2, 1985. Based on the earlier discussions, it is PGandE's understanding that these items will be included in the Combined Technical Specifications for Units 1 and 2 to be issued with the Unit 2 full power license.

1. T.S. 3.1.3.1, Movable Control Assemblies

PGandE License Amendment Request (LAR) 85-03, dated May 14, 1985, requested a change to Technical Specification 3.1.3.1, "Movable Control Assemblies, Group Height, Limiting Condition for Operation," to address multiple inoperable rods that are still trippable and also address multiple misaligned rods. This request was noticed in the Federal Register on June 18, 1985. The changes shown in Attachment A should be made for the Combined Units 1 and 2 Technical Specification 3.1.3.1, pages 3/4 1-15 and 1-16, when they are issued.

2. T.S. 3/4.3.1, Reactor Trip System Instrumentation

PGandE License Amendment Request (LAR) 85-04, dated May 20, 1985, requested a change to Technical Specification 3/4.3.1, "Reactor Trip System Instrumentation," to be consistent with the NRC Staff's position as given in Safety Evaluation Report, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," dated February 21, 1985. This request was noticed in the Federal Register on June 18, 1985. However, the results of discussions with the NRC Staff to resolve a point of clarification on the testing of one instrument were not included in the final draft Combined Technical Specifications. The approved clarification shown in Attachment A should be included in the Combined Technical Specification 3/4.3.1, pages 3/4 3-12 and 3/4 3-13, when they are issued.

3. T.S. 3.8.2.1 and 3.8.2.2, Electrical Power Systems

As agreed with the NRC Staff, the original issuance of the Combined Units 1 and 2 Technical Specifications should reflect the Unit 1 full power specifications as shown in Attachment A. The reasons for this are:

1. The Technical Specifications for Units 1 and 2 should be identical because the plant design which forms the bases for these specifications is identical.

2. Operation with the existing Unit 2 low power Technical Specifications could result in forcing the plant through unnecessary shutdowns and/or cooldowns in the event of a loss of a single dedicated battery charger. Redundant chargers and configurations were provided in the plant design to accommodate the loss of a single charger.

As a result of discussions with the NRC Staff regarding the Unit 1 and Unit 2 low power specifications, License Amendment Request (LAR) 85-07, dated May 31, 1985, requested a change to the Unit 1 specifications. The change would add an Action Statement to indicate that with more than one full-capacity charger receiving power simultaneously from a single 480 volt vital bus or any dc bus not receiving power from its associated ac division, the system would be restored to a configuration in which each charger is powered from its associated 480 volt vital bus within 14 days. The requested change was reviewed by the NRC Staff but was not prenoticed in the Federal Register. PGandE will submit a revision to LAR 85-07 to request this change to the Combined Units 1 and 2 Technical Specifications upon their issuance.

B. TYPOGRAPHICAL/EDITORIAL CHANGES

Corrections for typographical errors and minor editorial changes have been identified on the pages listed below and included in Attachment B:

vi, vii, viii, x, xi, xii, xiii	3/4 9-8, 9
xiv, xviii	3/4 11-2
1-1	3/4 12-4, 11
2-7, 9	B 3/4 0-1
3/4 1-5, 6, 14	B 3/4 1-1
3/4 2-7, 8, 9, 12, 17	B 3/4 2-2, 4, 5
3/4 3-11, 18, 21, 22	B 3/4 3-1, 3
3/4 3-34, 54, 56, 58	B 3/4 4-7, 8, 12, 13, 16
3/4 3-60, 66	B 3/4 6-1, 3
3/4 4-7, 13, 14	B 3/4 7-1, 6, 8
3/4 5-4, 11	B 3/4 8-2
3/4 6-2, 9, 17, 20	B 3/4 12-1
3/4 7-21, 28	5-3, 7
3/4 8-3, 7, 11, 20, 24, 25, 26	6-1, 6, 7

C. OTHER TECHNICAL SPECIFICATION ITEMS

1. T.S. 4.4.6.2.2, Table 3.4-1, Reactor Coolant System, Surveillance Requirements

Supplement 31 to the Safety Evaluation Report, Section 5.2.8.1(13) states that redundant isolation check valves in the safety injection system and the residual heat removal system shall be reclassified valve category A/C and tested in accordance with Technical Specification 4.4.6.2.2 and

included in Table 3.4-1 of the Technical Specifications. These check valves are:

8905A, SI to hot leg - 1	8905D, SI to hot leg - 4
8905B, SI to hot leg - 2	8740A, RHR to hot leg - 1
8905C, SI to hot leg - 3	8740B, RHR to hot leg - 2

The inclusion of these check valves in the Technical Specifications is still being discussed with the NRC Staff with resolution expected in the near future.

PGandE Letter No.: DCL-85-244

ATTACHMENT A TO ENCLOSURE 1

MARKED-UP TECHNICAL SPECIFICATION PAGES

TECHNICAL COMMENTS

1. Technical Specification 3.1.3.1, pages 3/4 1-15 and 3/4 1-16.
2. Technical Specification 3/4.3.1, pages 3/4 3-12 and 3/4 3-13.
3. Technical Specifications 3.8.2.1 and 3.8.2.2, pages 3/4 8-12 and 3/4 8-14.

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REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

~~b. With more than one full-length rod inoperable or misaligned from the group demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.~~

b. ~~c.~~ With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

1. The rod is restored to OPERABLE status within the above alignment requirements, or
2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or
3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:
    a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;
    b) THE SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

insert

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a above, POWER OPERATION may continue provided that:

1. Within one hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Figure 3.1-1a or Figure 3.1-1b, as applicable. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

d. With more than one rod misaligned from it's group ~~step counter~~ demand ~~height~~ position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
20. Reactor Trip System Interlocks (Continued)						
d. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	S/U(1)	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R	S/U(1, 8)	N.A.	N.A.	1
21. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
22. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
23. Seismic Trip	N.A.	R	N.A.	SA	R	1, 2

FINAL DRAFT

JUN 28 1985

FINAL DRAFT

TABLE 4.3-1 (Continued)

JUN 28 1985

TABLE NOTATIONS

* - When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

- Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

- Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) - Compare incore to excore axial flux difference above 15% of RATED THERMAL POWER at least once per 31 Effective Full Power days. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - Detector plateau curves shall be obtained and evaluated. For the Intermediate Range and Power Range Neutron Flux Channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) - Incore - Excore Calibration, above 75% of RATED THERMAL POWER at least once per 92 Effective Full Power days. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) - Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) - ~~Not used.~~ The surveillance requirement is not applicable for a reactor startup from MODES 2 or 3.

(9) - Quarterly Surveillance in MODES 3*, 4* and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(10) - Setpoint verification is not applicable.

(11) - At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include verification of the independence of the Undervoltage trip and the Shunt trip.

(12) - CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

(13) - Each channel shall be tested at least every 92 days on a STAGGERED TEST BASIS.

(14) - These channels also provide inputs to ESFAS. Comply with the applicable MODES and surveillance frequencies of Specification 4.3.2.1 for any portion of the channel required to be OPERABLE by Specification 3.3.2.

3/4.8.2 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following electrical busses shall be energized in the specified manner:

a. 4160 volt Vital Bus F,
b. 480 volt Vital Bus F,
c. 4160 volt Vital Bus G,
d. 480 volt Vital Bus G,
e. 4160 volt Vital Bus H,
f. 480 volt Vital Bus H,
g. 120 volt Vital Instrument A.C. Bus 1 energized from its associated inverter connected to D.C. Bus 1*,
h. 120 volt Supplemental Vital Instrument A.C. Bus 1A energized from its associated inverter connected to D.C. Bus 1*,
i. 120 volt Vital Instrument A.C. Bus 2 energized from its associated inverter connected to D.C. Bus 2*,
j. 120 volt Vital Instrument A.C. Bus 3 energized from its associated inverter connected to D.C. Bus 3*,
k. 120 volt Supplemental Vital Instrument A.C. Bus 3A energized from its associated inverter connected to D.C. Bus 3*,
l. 120 volt Vital Instrument A.C. Bus 4 energized from its associated inverter connected to D.C. Bus 2*,
m. 125 volt D.C. Bus 1 energized from Battery Bank 1, and ~~its~~ an associated full-capacity charger,
n. 125 volt D.C. Bus 2 energized from Battery Bank 2, and ~~its~~ an associated full-capacity charger, and
o. 125 volt D.C. Bus 3 energized from Battery Bank 3, and ~~its~~ an associated full-capacity charger.

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

ACTION:

a. With one of the required 4160 volt and/or associated 480 volt vital busses not energized, re-energize them within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one vital instrument A.C. bus not energized from its associated inverter, or with one inverter not connected to its associated D.C.

*Two vital instrument A.C. inverters or one vital and one supplemental vital instrument A.C. inverter may be disconnected from their D.C. busses for up to 24 hours for the purpose of performing an equalizing charge on their associated battery bank provided: (1) their vital busses are energized, and (2) the vital busses associated with the other battery banks are energized from their associated inverters and connected to their associated D.C. busses.

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LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One 4160-volt and its associated 480-volt A.C. vital bus,

b. Two 120-volt vital instrument A.C. busses and one 120-volt supplemental vital instrument A.C. bus energized from their associated inverters connected to their respective D.C. busses, and

c. One 125-volt D.C. bus energized from ~~its~~ on associated battery bank and full-capacity charger.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ATTACHMENT B TO ENCLOSURE 1

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BASES

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1.0 DEFINITIONS

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

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or trip setpoints such that the setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \leq \Delta T_0 \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)(T-T') + K_3(P-P') - f_1(\Delta I)\right]$$

align equal signs

Where:

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

T' = $\leq$ 576.6°F for Unit 1 and $\leq$ 577.6°F for Unit 2 Reference T_{avg} at RATED THERMAL POWER;

P = Pressurizer pressure, psig;

P' = 2235 psig (indicated RCS nominal operating pressure);

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 30 s, τ_2 = 4 s;

S = Laplace transform operator, s^{-1};

K_1 = 1.174;

K_2 = 0.01358/°F;

K_3 = 0.000685/psig;

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \leq \Delta T_0 [K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T - T'') - f_2(\Delta I)]$$

Where: ΔT_0 = Indicated ΔT at rated power;

T = Average temperature, °F;

T'' = $\leq$ 576.6°F for Unit 1 and $\leq$ 577.6°F for Unit 2 Reference T_{avg} at RATED THERMAL POWER;

K_4 = 1.079;

K_5 = 0.0174/°F for increasing average temperature and 0 for decreasing average temperature;

K_6 = 0.00121/°F for $T > T''$; $K_6 = 0$ for $T \leq T''$;

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation;

τ_3 = Time constant utilized in the rate lag controller for T_{avg} $\tau_3 = 10$ s;

S = Laplace transform operator, s^{-1}; and

$f_2(\Delta I)$ = $\boxed{0}$ for all ΔI.

align

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3%.

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SURVEILLANCE REQUIREMENTS ~~(Continued)~~

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a., above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to -3×10^{-4} $\Delta k/k/°F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3×10^{-4} $\Delta k/k/°F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 541°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore (T_{avg}) to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 551°F, with the T_{avg} - T_{ref} Deviation Alarm not reset.

#With K_{eff} greater than or equal to 1.

*See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS ~~(Continued)~~

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) Verifying the Boric Acid Storage System solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 35°F.

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SURVEILLANCE REQUIREMENTS ~~(Continued)~~

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER,

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in Specification 4.2.2.2b., above, to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f. below, and

2) The relationship:

$$F_{xy}^{L} = F_{xy}^{RTP}\ [1+0.2(1-P)],$$

Where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1) When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

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SURVEILLANCE REQUIREMENTS (Continued)

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.8.

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100% inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 for four loop operation.

Where: align

a. $R = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$,

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 include~~s~~ measurement uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2 and/or b., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 prior to exceeding the following THERMAL POWER levels:

   1. A nominal 50% of RATED THERMAL POWER,

   2. A nominal 75% of RATED THERMAL POWER, and

   3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 at least once per 12 hours when the value of R, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per 18 months.

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TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS
Actual Reactor Coolant System T_{avg}	$\leq$ 581°F (Unit 1) $\leq$ 582°F (Unit 2)
Actual Pressurizer Pressure	$\geq$ $\cancel{\leq}$ 2220 psia* (Unit 1) $\geq$ 2220 psia* (Unit 2)

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level-Low-Low	S	R	Q(13, 14)	N.A.	N.A.	1, 2
14. Steam Generator Water Level-Low Coincident with Steam/Feedwater Flow Mismatch	S	R	Q(13, 14)	N.A.	N.A.	1, 2
15. Undervoltage-Reactor Coolant Pumps	N.A.	R	N.A.	Q(13, 14)	N.A.	1
16. Underfrequency-Reactor Coolant Pumps	N.A.	R	N.A.	Q(13, 14)	N.A.	1
17. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	S/U(1, 10)	N.A.	1
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A	1, 2
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	R	N.A.	1
20. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	S/U(1)	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	S/U(1)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	S/U(1)	N.A.	N.A.	1

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					align
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
c. Containment Pressure-High-High	4	2	3	1, 2, 3	17
d. Steam Flow in Two Steam Lines-High	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, 3##	15*
Coincident With Either T_{avg}-Low-Low	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 3 loops	1, 2, 3##	15*
Or					
Steam Line Pressure-Low	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 3 loops	1, 2, 3##	15*
5. Turbine Trip & Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	25
b. Steam Generator Water Level-High-High	3/stm. gen.	2/stm. gen. in any operating stm. gen.	2/stm. gen. in each oper- stm. gen.	1, 2	15*

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TABLE 3.3-3 (Continued)

TABLE NOTATIONS

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

##Trip function may be blocked in this MODE below the P-12 (Low-Low T_{avg} Interlock) Setpoint.

subscript

*The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, declare the affected Emergency Diesel Generator(s) inoperable and comply with the ACTION statements of Specification 3.8.1.1; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 18 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves (RCV-11, 12, FCV 660, 661, 662, 663, 664) are maintained closed.

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TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 19 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 20 - 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:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement ~~s~~ is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 21 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 22 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 24 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated pump or valve inoperable and take the ACTION required by Specification 3.7.1.5 or 3.7.1.2 as applicable.

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines-High Coincident With Either	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
1) T_{avg}-Low-Low or	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2) Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
6. Auxiliary Feedwater								
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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INSTRUMENTATION

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent Chlorine Detection Systems,# with their Alarm/Trip Setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm, shall be OPERABLE.

APPLICABILITY: All MODES, when bulk chlorine gas is stored on the plant site.

ACTION:

a. With one Chlorine Detection System inoperable, restore the inoperable system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room Ventilation System in a recirculation mode with the HEPA filter and charcoal absorber system in operation.

b. With both Chlorine Detection Systems inoperable, within 1 hour initiate and maintain operation of the Control Room Ventilation System in a recirculation mode with the HEPA filter and charcoal absorber system in operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each Chlorine Detection System shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, and a CHANNEL FUNCTIONAL TEST at least once per 31 days. At least once per 18 months, the following inspections and maintenance shall be performed:

a. Check constant head bottle level and refill as necessary,

b. Clean the sensing cells,

c. Check flow meter operation and clean or replace filters and air lines as necessary,

d. Check air pump for proper operation, and

e. Verify that the detector responds to chlorine.

#The Chlorine Detection System is common to both units and is installed in the normal intakes to the Control Room Ventilation System.

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

PANEL A

ZONE	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE SMOKE	HEAT OR FLAME
1	Cable Spreading Room	10	4[1]
3	4KV Switchgear Bus F Room	1	N.A.
	4KV Switchgear Bus G Room	1	N.A.
	4KV Switchgear Bus H Room	1	N.A.
	4KV Switchgear Ventilation Fan Room	1	N.A.
4	4KV Switchgear Bus F Cable Spreading Area	1	N.A.
	4KV Switchgear Bus G Cable Spreading Area	1	N.A.
	4KV Switchgear Bus H Cable Spreading Area	1	N.A.
7	Containment Electrical Penetration Zone 7	4	N.A.
8	Containment Electrical Penetration Zone 8	3	N.A.
9	Rod Control Programmer/Reactor Trip Breaker Area	3	N.A.
	Battery No. 1 Charger Room	1	N.A.
	Battery No. 2 Charger Room	1	N.A.
	Battery No. 3 Charger Room	1	N.A.
10	480 Volt Bus F Area	1	N.A.
	480 Volt Bus G Area	1	N.A.
	480 Volt Bus H Area	1	N.A.
	Hot Shutdown Panel	1	N.A.
12	Inside Containment	17[2]	N.A.
13	Control Room Ventilation Return/Exhaust	1	N.A.
14	Control Room Ventilation Return/Exhaust	1	N.A.
15	Outside the Auxiliary Salt Water Pump Room	1	N.A.

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

PANEL B

ZONE	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE	
		SMOKE	HEAT OR FLAME
Not Assigned to Zone	Diesel Generator No. 1 Room	N.A.	$2^{(1)}$
	Diesel Generator No. 2 Room	N.A.	$2^{(1)}$
	Diesel Generator No. 3 Room(5)	N.A.	$2^{(1)}$
Not Assigned to Zone	Solid State Protection System Room	$3^{(4)}$	N.A.

TABLE NOTATIONS

(1) Heat sensors actuate CO_2 flooding and are tested per Specification 4.7.9.3c. Specifications 4.3.3.8.1 and 4.3.3.8.2 do not apply.

(2) The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

(3) Unit 1 Boric Acid Tank Detectors are in Zone 16, Unit 2.

(4) Smoke sensors actuate Halon flooding and are tested per Specification 4.7.9.4b. Specifications 4.3.3.8.1 and 4.3.3.8.2 do not apply.

(5) The fire pumps and Diesel Generator No. 3 are common to both units and are located on the Unit 1 side and on the Unit 1 Fire Detection Instrument Panel only.

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TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Line (RM-18)#	1	40
b. Steam Generator Blowdown Tank (RM-23)	1	41
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line (FR-20)#	1	43
b. Steam Generator Blowdown Effluent Lines (FR-53)	1	43
c. Oily Water Separator Effluent Line (FR-251)#	1	43
3. Radioactivity Monitor Not Providing Automatic Termination of Release		
Oily Water Separator Effluent Line (RM-3)#	1	42

#This Radioactive Liquid Effluent Monitoring Instrumentation is common to both units.

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TABLE 3.3-13 (Continued)

TABLE NOTATIONS

* At all times.

** During GASEOUS RADWASTE SYSTEM operation.

***MODES 1-4; also MODE 6 during CORE ALTERATIONS or movement of irradiated fuel within containment.

ACTION STATEMENTS

ACTION 50 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tanks contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 51 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 52 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 53 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend containment PURGING of radioactive effluents via this pathway.

ACTION 54 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 14 days. After 14 days or with no channels OPERABLE, operation of this system may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 55 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

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LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place a residual heat removal train into operation.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature~~s~~ and pressure.

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SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections not including the preservice inspection, result in all inspection results falling 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 may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a. The interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2; or

   2) A seismic occurrence greater than the Double Design Earthquake, or

   3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A main steam line or feedwater line break.

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SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1) 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;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

7) 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 a Double Design Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

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EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once each 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
8703	RHR to RCS Hot Legs	Closed
8802A	Safety Injection To RCS Hot Legs	Closed
8802B	Safety Injection to RCS Hot Legs	Closed
8809A	RHR to RCS Cold Legs	Open
8809B	RHR to RCS Cold Legs	Open
8835	Safety Injection to RCS Cold Legs	Open
8974A	Safety Injection Pump Recir. to RWST	Open
8974B	Safety Injection Pump Recir. to RWST	Open
8976	RWST to Safety Injection Pumps	Open
8980	RWST to RHR Pumps	Open
8982A	Containment Sump to RHR	Closed
8982B	Containment Sump to RHR	Closed
8992	Spray Additive Tank to Eductor	Open
8701	RHR Suction	Closed
8702	RHR Suction	Closed

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The Refueling Water Storage Tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 400,000 gallons,

b. A boron concentration of between 2000 and 2200 ppm ~~boron~~, and

c. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside ambient air temperature is less than 35°F.

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CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to L_a, 0.10 % ~~percent~~ by weight of the containment air per 24 hours at P_a, 47 psig, or, as applicable,

2) Less than or equal to L_t, 0.0472% by weight of the containment air per 24 hours at a reduced pressure of P_t, 25 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or the measured combined leakage rate for all penetrations and valves subject to Type~~s~~ B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations subject to Type B and C tests to less than or equal to 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
1. Phase "A" Isolation Valves		
FCV-151#	Steam Generator No. 1 Blowdown OC	$\leq$ 10
FCV-154#	Steam Generator No. 2 Blowdown OC	$\leq$ 10
FCV-157#	Steam Generator No. 3 Blowdown OC	$\leq$ 10
FCV-160#	Steam Generator No. 4 Blowdown OC	$\leq$ 10
FCV-244#	Steam Generator No. 4 Sample OC	$\leq$ 10
FCV-246#	Steam Generator No. 3 Sample OC	$\leq$ 10
FCV-248#	Steam Generator No. 2 Sample OC	$\leq$ 10
FCV-250#	Steam Generator No. 1 Sample OC	$\leq$ 10
FCV-253	Reactor Coolant Dr. Tk. PP Disch. Isol. IC	$\leq$ 10
FCV-254	Reactor Coolant Dr. Tk. PP Disch. OC	$\leq$ 10
FCV-255	Reactor Coolant Dr. Tk. Vent. Isol. IC	$\leq$ 10
FCV-256	Reactor Coolant Dr. Tk. Vent. Isol. OC	$\leq$ 10
FCV-257	Reactor Coolant Dr. Tk. Sample to GA OC	$\leq$ 10
FCV-258	Reactor Coolant Dr. Tk. Sample to GA IC	$\leq$ 10
FCV-260	Reactor Coolant Dr. Tk. N_2 Supply OC	$\leq$ 10
FCV-361	CCW Return from Excess Letdown HX OC	$\leq$ 10
FCV-500	Containment Sump Discharge Isolation IC	$\leq$ 10
FCV-501	Containment Sump Discharge Isolation OC	$\leq$ 10
FCV-584	Containment Instrument Air Supply OC	$\leq$ 10
FCV-633	Containment Fire Water Isolation OC	$\leq$ 10
FCV-654	Incore Cooler Chilled H_2O Supply OC	$\leq$ 10
FCV-655	Incore Cooler Chilled H_2O Supply IC	$\leq$ 10

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TABLE 3.6-1 (Continued)

VALVE NO.	FUNCTION	ISOLATION TIME (Seconds)
4. Manual Valves		
AIR-I-585*	Instrument Air Supply to Containment (FCV-584 Bypass) OC	N.A.
AIR-S-200*	Service Air Supply to Containment OC	N.A.
AXS-26*	Aux. Steam Supply to Containment OC	N.A.
CS-31	Containment Spray to Misc. Equipment Drain Tank OC	N.A.
CS-32	Containment Spray to Misc. Equipment Drain Tank OC	N.A.
FW-140#	Auxiliary Feedwater to Stm. Gen. No. 1 OC	N.A.
FW-147#	Auxiliary Feedwater to Stm. Gen. No. 2 OC	N.A.
FW-153#	Auxiliary Feedwater to Stm. Gen. No. 3 OC	N.A.
FW-157#	Auxiliary Feedwater to Stm. Gen. No. 4 OC	N.A.
MS-902#	Nitrogen to Steam Generators OC	N.A.
RCS-512*	Miscellaneous Equipment Drain Tank Isolation Valve OC	N.A.
SI-161*	Isolating Valve FI-927 OC	N.A.
VAC-1*	Containment Hydrogen Purge Supply Fan No. 1 and External H_2 Recombiner to Containment OC	N.A.
VAC-2*	Containment Hydrogen Purge Supply Fan No. 2 and External H_2 Recombiner to Containment OC	N.A.
8767	Refueling Cavity to Refueling Water Purification Pump OC	N.A.
8787	Refueling Water Purification Pump to Refueling Cavity OC	N.A.

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SURVEILLANCE REQUIREMENTS (Continued)

e. Functional Tests (Continued)

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location 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 test 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.

f. 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;

2) Snubber bleed, or release rate where required, is present in both tension~~s~~ and compression, within the specified range;

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions 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.

g. Functional Test Failure Analysis

An engineering 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.

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.9.2 The following Spray and/or Sprinkler Systems shall be OPERABLE:

a. 500/25 kV Main Transformer Bank,

b. 25/4 kV Unit Auxiliary Transformer No. 2,

c. 230/12 kV Standby Startup Transformer No. 1,

d. 12/4 kV Standby Startup Transformer No. 2,

e. Auxiliary Feedwater Pumps Area,

f. Centrifugal Charging Pumps Area,

g. Component Cooling Water Pumps Area, and

h. Containment Penetration Area.

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour, establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

b. At least once per 12 months by cycling each testable valve in the flow path through at least one cycle,

c. At least once per 18 months:

1) By performing a system functional test which includes simulated automatic actuation of the system, and:

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by:

1) Transferring 4 kV vital bus power supply from the normal circuit to the alternate circuit (manually and automatically) and to the delayed access circuit (manually), and

2) Verifying that on a Safety Injection test signal, without loss of offsite power, the preferred, immediate access offsite power source energizes the emergency busses with permanently connected loads and energizes the auto-connected emergency (accident) loads through sequencing timers.

4.8.1.1.2 Each diesel generator* shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:**

1) Verifying the fuel level in the engine-mounted fuel tank,

2) Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 13 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss of offsite power by itself (Startup bus undervoltage), or

c) A ~~Unit 2~~ Safety Injection actuation test signal by itself.

*Tests of Diesel Generator 3 to satisfy the frequency specified in Table 4.8-1 and in Surveillance Requirement 4.8.1.1.2.b for one unit may be counted in determining whether the frequency specified in Table 4.8-1 and in Surveillance Requirement 4.8.1.1.2b for the other unit is satisfied. Unit-specific portions of this Surveillance Requirement for Diesel Generator 3 shall be performed on an alternating schedule with signals from Units 1 and 2.

**All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Further, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures (e.g., gradual acceleration and/or gradual loading > 150 sec) as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

SURVEILLANCE REQUIREMENTS (Continued)

1) By verifying in accordance with the tests specified in ASTM-D975-81 prior to addition to the storage tanks that the sample has:

   a) An API Gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

   b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification;

   c) A flash point equal to or greater than 125°F; and

   d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.

2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.

d. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A;

e. At least once per 10 years by:

   1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and

   2) Performing a visual examination of accessible piping during an operating pressure leak test.

4.8.1.1.4 Reports - All diesel generator failures, valid or non-valid, shall be reported as a Special Report within 30 days to the Commission pursuant to Specification 6.9.2. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures (on a per nuclear unit basis) in the last 100 valid tests is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

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A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One diesel generator with:

1. An engine-mounted fuel tank containing a minimum volume of 200 gallons of fuel,

2. One supply train of the Diesel Fuel Oil Storage and Transfer system with storage of 8000 gallons of fuel in addition to the fuel required for the other unit.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel or crane operations with loads over the fuel storage pool. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2, 4.8.1.1.3, and 4.8.1.1.4, except for Specifications 4.8.1.1.1.b.2) and 4.8.1.1.2.a.2)c), b.2) for ESF times, b.6), b.7), b.10), and b.11).

TABLE 3.8-1

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
8107	Charging Isolation	Yes
8105	Charging Pumps' Recirculation	Yes
FCV-430	Component Cooling Heat Exch. Outlet	No
LCV-112B	Volume Control Tank Outlet	Yes
8801A	Boron Injection Tank Outlet	Yes
8803A	Boron Injection Tank Inlet	Yes
8807A	Safety Injection/Charging Suction Crosstie	No
8805A	RWST to Charging Pumps	Yes
FCV-437	Raw Water Supply to Auxiliary Feedpumps	No
FCV-441	S/G ~~SG~~ 4 Feedwater Isolation	Yes
FCV-750	RCP Thermal Barrier CCW Return	Yes
FCV-438	S/G ~~SG~~ 1 Feedwater Isolation	Yes
8923A	SI Pump 1 Suction	No
FCV-38	Aux. FWP Turb. Steam Supply	No
8980	RWST to RHR	No
8974A	SI Pumps' Recirculation	No
8992	Spray Additive Tank Outlet	Yes
8000A	Pressurizer RV Isolation	No
FCV-601*	Auxiliary Saltwater Pumps Crosstie	No
8808A	Accumulator No. 1 Isolation	No
8802A	SI Pump 1 to Hot Leg	No
8821A	SI Pump 1 to Cold Legs	No
8808D	Accumulator 4 Isolation	No
8808B	Accumulator 2 Isolation	No
8106	Charging Pumps' Recirculation	Yes
8108	Charging Isolation	Yes
LCV-112C	Volume Control Tank Isolation	Yes

*FCV-601 is common to both units.

SURVEILLANCE REQUIREMENTS (Continued)

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2) By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of "drawout" type circuit breakers shall consist of a CHANNEL CALIBRATION of the associated solid-state trip device for both the long-time delay trip element and the short-time delay element along with a breaker functional test. Testing of molded case circuit breakers shall consist of injecting a current with a value equal to 200% (for D.C. breakers) and 300% (for A.C. breakers) of the pickup of the time delay element and verifying that the circuit breaker operates within the time delay band for that current specified by the manufacturer. The instantaneous element of molded case circuit breakers shall be tested by injecting a current equal to -25%, +40% of the pickup value of the element and verifying that the circuit breaker trips with no intentional time delay. Circuit breakers found out-of-tolerance during functional testing shall be replaced prior to resuming operation. Circuit breakers that fail to trip magnetically before the withstand capability of the penetration conductor is reached shall be declared inoperable. Circuit breakers that fail to trip thermally before the manufacturer's maximum tolerance shall be declared inoperable. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested; and

3) By verifying that the thermal overload devices integral with the motor starters, used for penetration overcurrent protection, are OPERABLE by selecting a representative sample of at least 10% of the motor overload devices and performing a CHANNEL CALIBRATION. Motor overloads found inoperable shall be restored to OPERABLE status prior to resuming operation. For each motor overload device found inoperable, a CHANNEL CALIBRATION shall be performed on an additional representative sample of at least 10% of all the motor overload devices of the inoperable type until no more failures are found or a CHANNEL CALIBRATION has been performed on all motor overload devices of that type.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8-2
CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

UNIT 2*

PRIMARY DEVICE NO.	BACK-UP DEVICE NO.	EQUIPMENT DESCRIPTION
12 KV		
52VE7	52VE7R	Reactor Coolant Pump 1
52VD3	52VD3R	Reactor Coolant Pump 2
52VE6	52VE6R	Reactor Coolant Pump 3
52VD4	52VD4R	Reactor Coolant Pump 4
480 V "DRAWOUT"		
5223EI	5223EIR	Load Center 3I Feeder
5223DJ	5223DJR	Load Center 3J Feeder
480 V MCC		
522F02**	522F02R	Containment Fan Clr 1
522F01**	522F01R	Containment Fan Clr 2
522G01**	522G01R	Containment Fan Clr 3
522H01**	522H01R	Containment Fan Clr 4
522G02**	522G02R	Containment Fan Clr 5
522F40**	522F40R	PORV 8000A
522G46**	522G46R	PORV 8000B
522H33**	522H33R	PORV 8000C
522F46**	522F46R	Accum Inj to Cold Lp 8808A
522G07**	522G07R	Accum Inj to Cold Lp 8808B
522H14**	522H14R	Accum Inj to Cold Lp 8808C
522G05**	522G05R	Accum Inj to Cold Lp 8808D
522F23	522F23R	FCV750 RCP Barrier CCW Ret
522G38	522G38R	Emergency Ltg Xfmr
522G25**	522G25R	8701 RHR Suction
522H19**	522H19R	8702 RHR Suction
522G56**	522G56R	8703 RHR Recirc
522H18	522H18R	FCV749 RCP Brg Oil Clr
522H27	522H27R	8112 RCP Seal Water Ret
522F45	522F45R	Nuclear Rod Pos Racks
522G67	522G67R	H_2 Recomb
522H35	522H35R G	H_2 Recomb
522G37**	522H37R	FCV658 H_2 Recomb Iso
522H73**	522H73R	FCV659 H_2 Recomb Iso
5222J38**	5222J38R	CRDM Exh Fan 1
5222I40**	5222I40R	CRDM Exh Fan 2
5222J39**	5222J39R	CRDM Exh Fan 3
5222I39**	5222I39R	CRDM Exh Fan 4
5222I37**	5222I37R	RCWP Thrust Brg Lift Pump 1 RCP

*Unit 1 devices to be added before operation following the first refueling.

**These primary devices have motor overload relays that are an integral part of the motor starter as part of the primary protective device.

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TABLE 3.8-2 (continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

UNIT 2* PRIMARY DEVICE NO.	BACK-UP DEVICE NO.	EQUIPMENT DESCRIPTION
5222J13**	5222J13R	RCWP Thrust Brg Lift Pump 2
5222I21**	5222I21R	RCWP Thrust Brg Lift Pump 3
5222J18**	5222J18R	RCWP Thrust Brg Lift Pump 4
5222I10**	5222I10R	Containment Sump Pump 1
5222J10**	5222J10R	Containment Sump Pump 2
5222I30**	5222I30R	Containment Sump Pump 3
5222J29**	5222J29R	Containment Sump Pump 4
5222I31**	5222I31R	Reactor Clnt Drain Pump 1
5222J07**	5222J07R	Reactor Clnt Drain Pump 2
5222I26**	5222I26R	Iodine Rem Exh Fan 5
5222J02**	5222J02R	Iodine Rem Exh Fan 6
5222I25**	5222I25R	Reactor Cavity Sump Pump 1
5222J11**	5222J11R	Reactor Cavity Sump Pump 2
5222I08	5222I08R	Containment Sup Fan 6
480 V HEATER PANELS		
52PH211	52PH211R	Pressurizer Heater Panel 1
52PH212	52PH212R	Pressurizer Heater Panel 1
52PH213	52PH213R	Pressurizer Heater Panel 1
52PH214	52PH214R	Pressurizer Heater Panel 1
52PH215	52PH215R	Pressurizer Heater Panel 1
52PH216	52PH216R	Pressurizer Heater Panel 1
52PH221	52PH221R	Pressurizer Heater Panel 2
52PH222	52PH222R	Pressurizer Heater Panel 2
52PH223	52PH223R	Pressurizer Heater Panel 2
52PH224	52PH224R	Pressurizer Heater Panel 2
52PH225	52PH225R	Pressurizer Heater Panel 2
52PH226	52PH226R	Pressurizer Heater Panel 2
52PH227	52PH227R	Pressurizer Heater Panel 2
52PH231	52PH231R	Pressurizer Heater Panel 3
52PH232	52PH232R	Pressurizer Heater Panel 3
52PH233	52PH233R	Pressurizer Heater Panel 3
52PH234	52PH234R	Pressurizer Heater Panel 3
52PH235	52PH235R	Pressurizer Heater Panel 3
52PH236	52PH236R	Pressurizer Heater Panel 3
52PH237	52PH237R	Pressurizer Heater Panel 3
52PH241	52PH241R	Pressurizer Heater Panel 4
52PH242	52PH242R	Pressurizer Heater Panel 4
52PH243	52PH243R	Pressurizer Heater Panel 4
52PH244	52PH244R	Pressurizer Heater Panel 4
52PH245	52PH245R	Pressurizer Heater Panel 4
52PH246	52PH246R	Pressurizer Heater Panel 4

*Unit 1 devices to be added before operation following the first refueling.
**These primary devices have motor overload relays that are an integral part of the motor starter as part of the primary protective device.

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3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) train shall be OPERABLE and in operation.***

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is at least 23 feet.

ACTION:

With no RHR train OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR train to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR train shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

*The RHR train may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

**The RHR train may be removed from operation and OPERABLE status for up to 2 hours per 8 hour period for the performance of leak testing the RHR suction isolation valves.

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REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) trains shall be OPERABLE and at least one RHR train shall be in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR trains OPERABLE, immediately initiate corrective action to return the required RHR trains to OPERABLE status, or to establish at least 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR train 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 RHR train to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR train shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

*Prior to initial criticality, the RHR train may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

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TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD) (µCi/ml)(1)
1. Batch Waste Release Tanks(4)	P Each Batch	P Each Batch	Principal Gamma Emitters(6)	$5x10^{-7}$
			I-131	$1x10^{-6}$
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	$1x10^{-5}$
	P Each Batch	M Composite(2)	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	P Each Batch	Q Composite(2)	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
2. Continuous Releases(5) Steam Generator Blowdown Tank	D Grab Sample	W Composite(3)	Principal Gamma Emitters(6)	$5x10^{-7}$
			I-131	$1x10^{-6}$
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	D Grab Sample	M Composite(3)	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	D Grab Sample	Q Composite(3)	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
3. Continuous Releases(5) Oily Water Separator Effluent	D Grab Sample	W Composite(3)	Principal Gamma Emitters(6)	$5x10^{-7}$

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne			
Radioiodine and Particulates	Samples from six locations (MT1, OS2, 8S1, 2F2, 5F1 and 7D1);	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Canister: I-131 analysis weekly.
	Three samples (MT1, OS2 and 8S1) from close to the three SITE BOUNDARY locations, in different sectors, two in sectors with the highest calculated annual average ground-level D/Q, the other in the South sector;		Particulate Sampler: Gross beta radioactivity analysis following filter change;(3) and gamma isotopic analysis(4) of composite (by location) quarterly.
	One sample (7D1) from the vicinity of a community having the highest calculated annual average ground-level D/Q;		
	One sample (5F1) from the San Luis Obispo area; and		
	One sample (2F2) from a control location.		

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TABLE 4.12-1 (Continued)

TABLE NOTATIONS (Continued)

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

3/4.0 APPLICABILITY

BASES

The specifications for this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4. In the event of a disagreement between the requirements stated in these Technical Specifications and those stated in an applicable Federal Regulation or Act, the requirements stated in the applicable Federal Regulation or Act shall take precedence and shall be met except where a specific exemption to the regulation has been granted in the Technical Specifications or Facility License.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 The specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.5.2 requires two ECCS subsystems to be OPERABLE and provides explicit ACTION requirements if one ECCS subsystem is inoperable. Under the requirements of Specification 3.0.3, if both the required ECCS subsystems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one Spray System is inoperable. Under the requirements of Specification 3.0.3 if both the required Containment Spray Systems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN within the subsequent 24 hours. It is acceptable to initiate and complete a reduction in OPERATIONAL MODES in a shorter time interval than required in the ACTION statement and to add the unused portion of this allowable out-of-service time to that provided for operation in subsequent lower OPERATIONAL MODE(S). Stated allowable out-of-service times are applicable regardless of the OPERATIONAL MODE(S) in which the inoperability is discovered but the times provided for achieving a mode reduction are not applicable if the inoperability is discovered in a mode lower than the applicable mode. For example if the Containment Spray System was discovered to be inoperable while in STARTUP, the ACTION Statement would allow up to 156 hours to achieve COLD SHUTDOWN. If HOT STANDBY is attained in 16 hours rather than the allowed 78 hours, 140 hours would still be available before the plant would be required to be in COLD SHUTDOWN. However, if this system was discovered to be inoperable while in HOT STANDBY, the 6 hours provided to achieve HOT STANDBY would not be additive to the time available to achieve COLD SHUTDOWN so that the total allowable time is reduced from 156 hours to 150 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with: (1) the full complement of required systems, equipment, or components OPERABLE and (2) all

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.6% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the FSAR accident and transient analysis.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC) was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition, and a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -3.9×10^{-4} $\Delta k/k/°F$. The MTC value of -3.0×10^{-4} $\Delta k/k/°F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -3.9×10^{-4} $\Delta k/k/°F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of each fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the ±5% target band about the target flux difference, during rapid plant THERMAL POWER changes, control rod motion will cause the AFD to deviate outside of the target band. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached subsequently (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for 2 or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on Heat Flux Hot Channel Factor, RCS Flowrate, and Nuclear Enthalpy Rise Hot Channel Factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded, and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ~~i~~ensure that the limits are maintained provided:

1. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position,

2. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6,

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

3. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained, and

4. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions 1. through 4., above, are maintained. As noted on Figure 3.2-3a and Figure 3.2-3b, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R, as calculated per Specification 3.2.3 and used in Figure 3.2-3a and Figure 3.2-3b accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is the value used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed. Thus, knowing the "as measured" values of $F^N_{\Delta H}$ and RCS flow allows for "trade offs" in excess of R equal to 1 for the purpose of offsetting the rod bow DNBR penalty.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margin totaling 9.1% DNBR is derived from the difference between the design and required values on the following items: (a) design DNBR limit, (b) grid spacing multiplier, (c) thermal diffusion coefficient, (d) DNBR spacer factor multiplier and (e) pitch reduction. The rod bow penalty is calculated with the method described in WCAP-8691, Revision 1, and is completely compensated by the available margin of 9.1%.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3. Measurement errors of 3.5% for RCS total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3a and Figure 3.2-3b.

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide additional assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.8 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3% for each percent of tilt in excess of 1.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and safety analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

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3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and Engineered Safety Features Actuation System instrumentation and interlocks ensure that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation, and (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functinal capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out-of-service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those engineered safety features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident: (1) safety injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment ~~cooling fans~~ start and automatic valves position, and (11) component cooling water pumps start and automatic valves position.

fan cooler units

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of ~~Regulatory~~ Safety Guide ~~1~~ 12, "Instrumentation for Earthquakes," ~~April 1974.~~

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as ~~a~~ result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. The normal plant instrument channels specified are suitable for use as post-accident instruments. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983, and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection System ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Presure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 560°F, and

5. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness testing of the ferritic materials in the reactor vessel ~~were~~ was performed in accordance with the 1966 Edition for Unit 1 and the 1968 Edition for Unit 2 of the ASME Boiler and Pressure Vessel Code, Section III. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 6 effective full power years (EFPY) of service life. The 6 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region

TABLE B 3/4.4-1a

REACTOR VESSEL TOUGHNESS DATA-UNIT 1

COMPONENT lowercase	PLATE NO.	MATERIAL TYPE	Cu (Wt%)	P (Wt%)	NDTT °F	MINIMUM 50 FT-LB/35 Mil TEMP °F LONG	MINIMUM 50 FT-LB/35 Mil TEMP °F TRANS	RT_{NDT} °FT	AVERAGE UPPER SHELF FT-LB LONG	AVERAGE UPPER SHELF FT-LB TRANS
CL.HD.DOME	B4108	A533B1			-30	51	71*	11		72*
CL.HD.SEG.	B4109-1	A533B1			0	53	73*	13		81*
CL.HD.SEG.	B4109-2	A533B1			-10	50	70*	10		89*
CL.HD.SEG.	B4109-3	A533B1			-20	50	70*	10		79*
CL.HD.FLG.	B4102	A508,2			53*	30	50*	53		103*
VES.SH.FLG.	B4101	A508,2			35*	-5	15*	35		99*
INLET NOZ.	B4103-1	A508,2			60*	17	37*	60		77*
INLET NOZ.	B4103-2	A508,2			60*	27	47*	60		75*
INLET NOZ.	B4103-3	A508,2			43*	10	30*	43		108*
INLET NOZ.	B4103-4	A508,2			48*	2	22*	48		106*
OUTLET NOZ.	B4104-1	A508,2			60*	-13	7*	60		77*
OUTLET NOZ.	B4104-2	A508,2			43*	-3	17*	43		74*
OUTLET NOZ.	B4104-3	A508,2			54*	-12	8*	54		86*
OUTLET NOZ.	B4104-4	A508,2			60*	30	50*	60		84*
UPPER SHL.	B4105-1	A533B1	0.12	0.010	10	68	88*	28		80*
UPPER SHL.	B4105-2	A533B1	0.12	0.008	0	~~48~~ 49	69*	9		74*
UPPER SHL.	B4105-3	A533B1	0.14	0.010	0	54	74*	14		81*
INTER.SHL.	B5106-1	A533B1	0.14	0.013	-10	57	40	-10	134	116
INTER.SHL.	B4106-2	A533B1	0.13	0.013	-10	36	57	- 3	132	114
INTER.SHL.	B4106-3	A533B1	0.10	0.011	10	70	90*	30	119	77.5*
LOWER SHL.	B4107-1	A533B1	0.13	0.011	-10	59	75	15	127	110
LOWER SHL.	B4107-2	A533B1	0.12	0.010	-10	64	80	20	127	108
LOWER SHL.	B4107-3	A533B1	0.12	0.010	-50	52	38	-22	135	116
BOT.HD.SEG.	B4111-1	A533B1			-20	33	53*	- 7		82*
BOT.HD.SEG.	B4111-2	A533B1			-40	16	36*	-14		90*
BOT.HD.SEG.	B4111-3	A533B1			-40	21	41*	-19		85*
BOT.HD.SEG.	B4110	A553B1			-10	60	80*	20		75*

*Estimated per NRC Standard Review Plan Section 5.3.2.

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PRESSURE/TEMPERATURE LIMITS (Continued)

is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1a for Unit 1 and Table B 3/4.4-1b for Unit 2. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content and phosphorous content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 6 EFPY.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185-73, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. The heatup and cooldown curves must be recalculated ~~with~~ when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in the following paragraphs.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the

PRESSURE/TEMPERATURE LIMITS (Continued)

calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\,K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where, K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

align

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature of the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

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LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

The Maximum Allowed PORV Setpoint for the LTOPs will be modified, if required, based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for the ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.11 REACTOR VESSEL HEAD VENTS

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

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

The function, capabilities, and testing requirements of the Reactor Coolant System Vent Systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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BASES

3/4.6.1 CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates ~~are~~ is consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 47 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 46.91 psig. This includes the limit of 0.3 psig for initial positive containment pressure. The total pressure is less than design pressure and is consistent with the safety analyses.

CONTAINMENT SYSTEMS

FINAL DRAFT
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BASES

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration ensure that: (1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and (2) corrosion effects on components within containment are minimized. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment fan cooler units ensures that: (1) the containment air temperature will be maintained within limits during normal operation, (2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post LOCA conditions, and (3) adequate mixing of the containment atmosphere following a LOCA to prevent localized accumulations of hydrogen from exceeding the flammable limit.

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out of service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

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BASES

for Unit 1, and 110 % of the total secondary steam flow of 1.49 x 10^7 lb/hr at 100% RATED THERMAL POWER for Unit 2.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to not more than 105% of its design pressure of 1065 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1968 Edition. The total relieving capacity for all valves on all of the steam lines is 1.645 x 10^7 lb/hr which is 113% of the total secondary steam flow of 1.45 x 10^7 lb/hr at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

109 = Power Range Neutron Flux-High Trip Setpoint,

X = Total relieving capacity of all safety valves per steam line in lb/hour, and

Y = Maximum relieving capacity of any one safety valve in lb/hour.

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the Fire Suppression Systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The Fire Suppression System consists of the water system, spray and/or sprinklers, CO_2, Halon, and fire water hose stations. The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility Fire Protection Program.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying the weight of the tanks.

In the event the Fire Suppression Water System becomes inoperable, prompt corrective measures must be taken since this system provides the major fire suppression capability of the plant.

3/4.7.13 FLOOD PROTECTION

The breakwaters (east and west) provide flood protection to the safety-related auxiliary salt water (ASW) pumps located in the intake structure. The ASW pumps would be flood protected for the events up to and including the probable maximum tsunami and maximum credible wave, if the breakwaters are OPERABLE at a level of 0 feet mean lower low water (MLLW) or above. However, substantial degradation of the breakwaters below MLLW level may result in the ASW pump being flooded under the design basis flood and would require corrective actions to be taken. The ongoing surveillance of the breakwaters will ensure that proper flood protection is provided for the ASW pumps.

BASES

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

from one unit is maintained during surveillance testing on the other unit, then the third (common) Diesel Generator Unit shall be considered to be OPERABLE for that unit.

The Surveillance Requirements for demonstrating the OPERABILITY of the batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-3 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-3 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the Radiological Effluent Monitoring Program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in Currie, L.A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program given in the ERMP are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

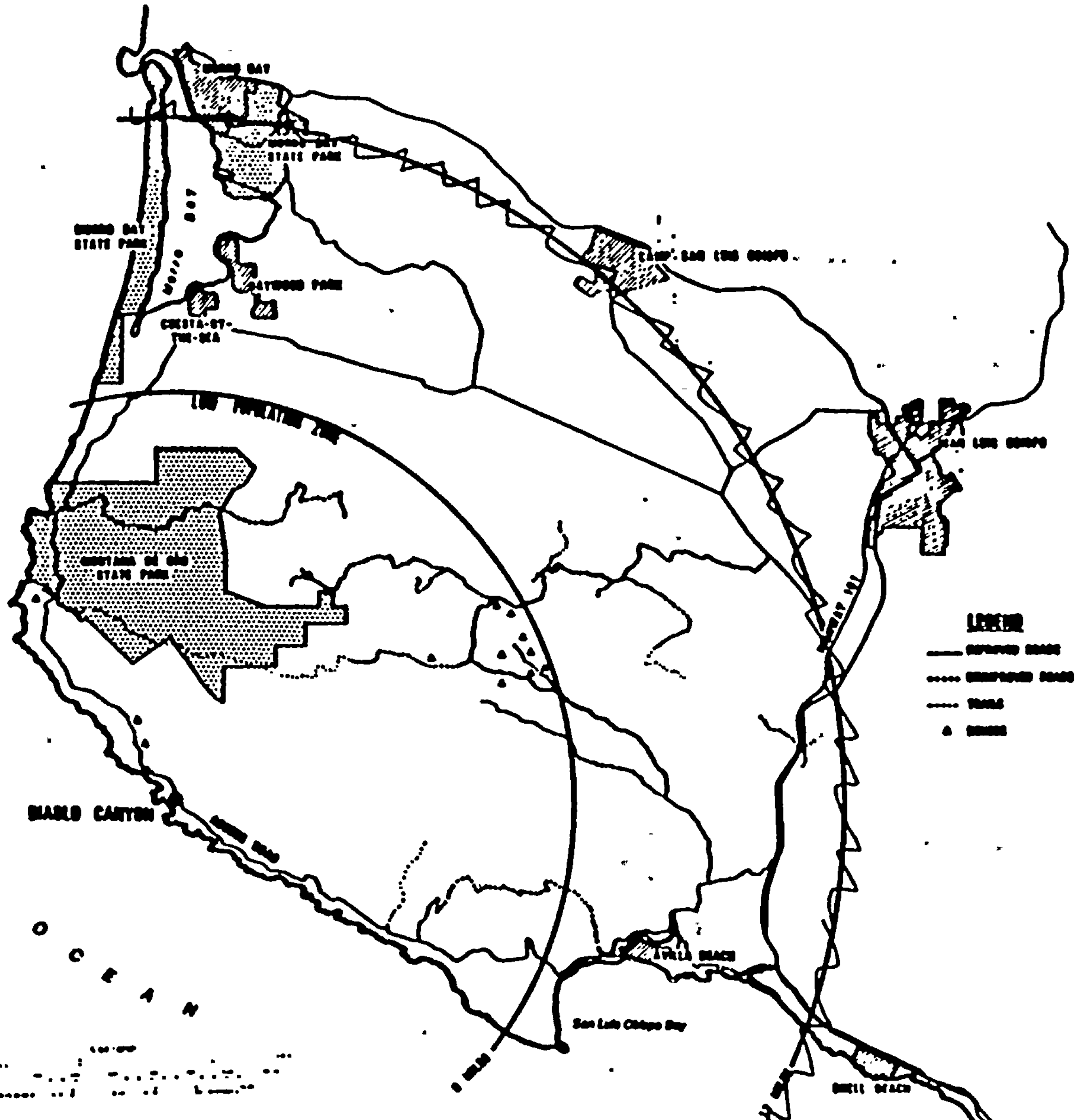

FIGURE 5.1-2

LOW POPULATION ZONE

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TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	250 Heatup and Cooldown Cycles	200°F to 550°F to 200°F except for pressurizer, which is 200°F to 650°F to 200°F
	100 Loss of load cycles, with turbine trip and without immediate reactor trip	Above 15% RTP to 0% RTP
	50 cycles of loss of all offsite electrical power	Loss of all offsite power with turbine trip and initial power level at 100% RTP
	100 cycles of loss of flow in one reactor coolant loop	Loss of one reactor coolant pump with initial power level at 100% RTP
	500 reactor trip cycles	100% to 0% RTP
	12 inadvertent auxiliary spray actuation cycles	Spray water temperature differential > 320°F and < 560°F
	60 leak tests	Pressurized to ≥ 2500 psig coincident with no pressurization of the secondary side of steam generators
	10 hydrostatic pressure tests	Pressurized to ≥ 3107 psig
Secondary Coolant System	10 hydrostatic pressure tests each steam generator	Pressurized to ≥ 1356 psig coincident with the primary side at 0 psig
	10 Turbine roll tests	Turbine roll on RCP heat resulting in plant cooldown rate > 100°F/hr

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6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall plant operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room Command function. A management directive to this effect signed by the Vice President, Nuclear Power Generation shall be reissued to all plant personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for plant management and technical support shall be as shown on Figure 6.2-1.

PLANT STAFF

6.2.2 The plant organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Operator shall be in the Control Room;

c. A Health Physics Technician* shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site Fire Brigade of at least five members* shall be maintained onsite at all times. The Fire Brigade shall not include the Shift Supervisor and the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

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*The Health Physics Technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

COMPOSITION

6.2.3.2 The OSRG shall be composed of at least five engineers located on site.

RESPONSIBILITIES

6.2.3.3 The OSRG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4~~6~~ The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the plant.

6.3 PLANT STAFF QUALIFICATIONS

6.3~~6~~ Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Chemistry and Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 for Radiation Protection Manager. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

*Not responsible for sign-off function.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of a designated member of the facility staff and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT STAFF REVIEW COMMITTEE (PSRC)

FUNCTION

6.5.1.1 The Plant Staff Review Committee shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Staff Review Committee shall be composed of the:

Chairman:	Plant Manager
Member:	Plant Superintendent
Member:	Operations Manager
Member:	Assistant Plant Manager, Support Services
Member:	Quality Control Manager
Member:	Maintenance Manager
Member:	Assistant Plant Manager, Technical Services
Member:	Chemistry and Radiation Protection Manager

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PSRC Chairman to serve on a temporary basis; however, no more than one alternate shall participate as a voting member in PSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PSRC shall meet at least once per calendar month and as convened by the PSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the PSRC necessary for the performance of the PSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and two members one of which may be an alternate.

ENCLOSURE 2

DIABLO CANYON UNITS 1 AND 2 COMBINED TECHNICAL SPECIFICATION DEVELOPMENT AND REVIEW PROCESS

PGandE's development and review process for the Diablo Canyon Unit 1 Technical Specifications (NUREG-1102) and Diablo Canyon Unit 2 low power Technical Specifications (NUREG-1132) was described in detail in Enclosure 2 of PGandE letter DCL-85-167, dated April 25, 1985 which provided certification of the Unit 2 low power Technical Specifications. Enclosures 1 and 2 to the April 25, 1985 letter also indicated that discussions were held with the NRC Staff regarding preparation of the Combined Units 1 and 2 Technical Specifications and that certain outstanding issues were still being discussed for resolution prior to issuance of the Combined Technical Specifications. The following paragraphs describe PGandE's actions for development and review of the Combined Technical Specifications since issuance of the Unit 2 low power Technical Specifications.

Seventeen Technical Specification items for the full power license were still outstanding when the Unit 2 low power license was issued. These items were discussed with the NRC Staff in a meeting on May 15, 1985 and were identified in an NRC letter to PGandE which indicated that the 17 items should be resolved for the Combined Units 1 and 2 full power Technical Specifications. On May 21, 1985, PGandE responded to the 17 items and indicated that a License Amendment Request (LAR) would be submitted for seven items that required a Technical Specification change. LARs 85-05 and 85-07 were submitted on May 30, 1985 and May 31, 1985 respectively to address these items.

Eight of the items were resolved with the Staff with no further action required. The remaining two items regarding actions required for inoperable PORVs and inclusion of the loose-part detection system were resolved with the NRC Staff and a commitment was made in a PGandE letter dated June 20, 1985 to provide an LAR to revise Combined Units 1 and 2 Technical Specification 3.4.4, "Reactor Coolant System, Relief Valves, Limiting Condition for Operation," and include a specification for "Instrumentation Loose-Part Detection System" within 90 days of receipt of the Unit 2 full power license. The June 20, 1985 letter also included a commitment to provide LAR(s) to make minor changes to: 1) administrative controls for startup reports, 2) bases for electrical power system, and 3) reactor coolant system pressure/temperature limits figures.

In addition to PGandE personnel meeting several times with the NRC Staff and submitting letters and LARs regarding the Combined Units 1 and 2 Technical Specifications, PGandE performed its review of the proposed final draft combined full power specifications by comparing them to the Unit 2 low power Technical Specifications, LAR 85-01 and its revisions and LAR 85-03 through LAR 85-06, and the seventeen items discussed above for full power Combined Units 1 and 2 Technical Specifications. These LARs were approved and noticed in the Federal Register by the NRC Staff. Those Technical Specifications which are unit-specific were also reviewed to ensure applicability to that unit. The final certification process by PGandE included page-by-page,

line-by-line review of the final draft Combined Units 1 and 2 full power Technical Specifications against the specifications which PGandE believed had been established through discussions with the NRC Staff.

The final review did not include a detailed in-depth technical review, as this type of review had been performed over the past weeks and years, but consisted of a comparison to determine that the specifications provided for review reflected the agreement reached between PGandE and the NRC concerning the composition of these specifications.

Resources employed to perform these reviews included experienced PGandE managers, engineers, and included several personnel holding operators licenses and vendor representatives.