Georgia Power Company
Route 2, Box 299A
Waynesboro, Georgia 30830
Telephone 404 554-9961
404 724-8114

Southern Company Services, Inc.
Post Office Box 2625
Birmingham, Alabama 35202
Telephone 205 870-6011

Vogtle Project

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CONSTRUCTION PERMIT NUMBER CPPR-109
VOGTLE ELECTRIC GENERATING PLANT - UNIT 2
TECHNICAL SPECIFICATIONS

Gentlemen:

Enclosed for your staff's review are draft Technical Specifications applicable to Vogtle Electric Generating Plant - Unit 2. The draft Technical Specifications are based on the current Unit 1 Technical Specifications, and are prepared in such a way as to apply to both units, thus they are designated as Vogtle Electric Generating Plant Units 1 and 2 Technical Specifications.

This draft retains the current requirements for Unit 1 and applies consistent Technical Specification requirements to Unit 2.

The changes shown on the enclosure consist of (1) format or editorial changes necessary to reflect the operation of Unit 2, (2) requirements or exceptions that are unique to Unit 2, (3) changes that are necessary to reflect differences between Unit 1 and 2 design, and (4) changes which are being requested for Unit 1 and which are expected to be approved prior to Unit 2 licensing. In addition, some minor editorial or format changes that are not due to the operation of Unit 2 have been included. These draft Technical Specifications are presented as a marked-up version of the current Unit 1 Technical Specifications such that changes are easily noted.

A description and justification for each change has been provided on a separate page. An index is provided which correlates the descriptions and justifications to the affected pages. The index, descriptions and justifications follow the title page.

Because the Unit 1 and 2 design is similar, the required Technical Specification changes are not extensive. Some changes could not be included because analyses, preoperational tests, or design details have not been finalized. Where information is to be provided when available, the particular specification has been annotated. Examples of these types of changes include (1) changes to the instrumentation tables, chlorine monitor specification and Control Room Emergency Filtration specification to reflect the combined control room operation, and (2) changes to the heatup and cooldown curves for Figures 3.4-2 and 3.4-3 for Unit 2 operation. In addition, changes to Chapter 16 of the FSAR to reflect Unit 2 will be provided by separate letter.

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It is anticipated that additional changes to the Unit 1 Technical Specifications will be requested based on operational experience and Westinghouse Owner's Group efforts. Concurrent with any requests for changes to the Unit 1 Technical Specifications, a request to change the draft Units 1 and 2 Technical Specifications will be made while the draft Units 1 and 2 Technical Specifications are under review.

It is intended that the Technical Specifications as referenced in the Unit 2 license will reflect all of the Unit 1 Technical Specifications in effect at the time the Unit 2 license is issued. At that time Georgia Power will request an amendment to the Unit 1 license to incorporate the Unit 1 and 2 Technical Specifications. This will provide a single Technical Specification document applicable to both units.

An original plus four copies is being provided. If your staff requires additional information, please do not hesitate to contact me.

Sincerely,

J. A. Bailey

J. A. Bailey
Project Licensing Manager

JAB/HWM/lg
Attachment
xc: NRC Regional Administrator
NRC Resident Inspector
J. P. O'Reilly
P. D. Rice
L. T. Gucwa
R. A. Thomas
J. E. Joiner, Esquire
J. B. Hopkins (2)
G. Bockhold, Jr.
R. Goddard, Esquire
R. W. McManus
Vogtle Project File

1386V

TECHNICAL SPECIFICATIONS
Vogtle Electric Generating Plant,
Units 1 and 2
Dockets 50-424 and 50-425

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1. Description of the Change

Page	Typographical, Grammatical and Editorial Changes
VI	Change "Loose-Parts" to "Loose Parts" within index item Table 3.3-8
XXIII	Change "Reports" to "Report" within index item 6.8.1
2-1	Change "parenthesis" to "parentheses" in the footnote
B 2-7	Add a comma after power in the first line of the last paragraph
B 2-8	Add a comma after power in the first line of the first paragraph
3/4 0-1	Change "parenthesis" to "parentheses" in the footnote
3/4 1-3	Change "MODE" to "MODES" in the APPLICABILITY statement, change "the" to "an" and change "rod" to "rod(s)" in Specification 4.1.1.2a
3/4 3-23	Change "P11" to "P-11" in item 9.a. of Table 3.3-2
3/4 3-34	Delete the word "system" from item 11 of Table 3.3-3
3/4 3-48	Change "RE-121117" to "RE-12117"
3/4 3-50	Add a space between the 1 and the g of 0.01 g in 4.3.3.3.2
3/4 3-64	Change 3/4.3.3.3.8 to 3/4.3.3.8
3/4 3-80	Change "modes" to "MODES" in the footnote
3/4 5-3	Change "Mode" to "MODE" in the footnote
3/4 6-8	Change "accceptable" to "acceptable" in 4.6.1.6.1.a
3/4 6-9	Change "not" to "no" in 4.6.1.6.1.b.2
3/4 7-4	Change "FI-5100" to "FI-15100" in 4.7.1.2.1.a.2)
3/4 7-12	Change "Service water system " to "NSCW" in 4.7.4.b.2
3/4 8-17	Add "MCC" after "switchgear" in ACTION c.
3/4 8-18	Change "RCS" to "Reactor Coolant System" in the ACTION Statement
B 3/4 1-1	Add a comma after the number 2 and the number 5 in the second paragraph of Bases 3/4.1.1.1, and add the word "required" in the last sentence of the paragraph
B 3/4 2-2	Add at the end of the title of 3/4.2.2
B 3/4 3-5	Change "Part" to "Parts" in the title for 3/4.3.3.8
B 3/4 7-1	Indent the word POWER in the definition of SP
6-12	Delete word "and" at the end of item 1
6-17	Add "The" as the first word of the second sentence of 6.8.1.2a
6-18	Change "reports" to "report" in 4 places in the first 3 paragraphs of this page

These changes do not affect any of the current Unit 1 Technical Specification requirements, although they are equally applicable to the Unit 1 Technical Specifications.

Justification

These are corrections of spelling, syntax, and format, that are not specifically required to show Unit 2 requirements and do not change any requirement of Unit 1 or Unit 2. In some cases more acceptable nomenclature is used for the same item, such as replacing Service Water System by NSCW. The affected pages are given in the Description of the Change.

2. Description of the Change

Footnote * is being revised for Section 2.0, Safety Limits and Limiting Safety System Settings and 3/4.0, Limiting Conditions for Operation and Surveillance Requirements, and a new footnote ** is being added to 3/4.0. These footnotes do not change any requirements but provide clarification concerning the applicability of the Technical Specifications to both units. They explain that information such as instrument numbers, numbers of components, or numbers of required channels, trains, components etc. as well as required actions are applicable to each unit except where specifically noted otherwise.

This change does not affect the current Unit 1 requirements.

Justification for the Change

Most of the information contained in the Technical Specifications is on a per unit basis, for example a specified setpoint is the same for the Unit 1 instrument as for the Unit 2 instrument, or the required number of channels is the same for Unit 1 as for Unit 2 or the instrument number is the same for Unit 1 as for Unit 2. Where exceptions exist they will be specifically noted in the individual specification by pointing out that the requirement is for Unit 1, Unit 2 specifically, or common to both units.

Pages affected.
Page 2-1
Page 3/4 0-1

3. <u>Description of the Change</u>

The trip setpoint of the Steam Generator Water Level Low-Low is specified at _ 18.5% of the narrow range span with a footnote that requires it to be set at _ 22.5% until resolution of the Veritrak uncertainty issue. The applicable value to be specified in the footnote for Unit 2 is 23.5%. This will also be reflected in Table Notation # of Table 3.3-3.

This change does not affect the current Unit 1 requirements.

<u>Justification for the Change</u>

Veritrak transmitter uncertainties in excess of those identified in the Equipment Specification and assumed in the safety analysis have been discovered in transmitters manufactured by Veritrak.

Unit 1 and 2 Narrow Range Steam Generator Water Level Low-Low Signal Environmental Allowances were found to be outside the standard 10% value. A higher value of Environmental Allowance for Unit 2 results in a different setpoint for Unit 2, of 23.5%, for the Steam Generator Water Level Low-Low signal for Reactor Trip and Auxiliary Feedwater initiation.

Pages affected.

Page 2-5, Table 2.2-1
Page 3/4 3-35, Table 3.3-3

4. Description of the Change

Footnotes were added in the following Tables' Functional Units (Items)/Monitors to reference their inclusion in the Accident Monitoring Instrumentation Specification (3/4.3.3.6). When required the footnote which references Specification 3/4.3.3.6 was added.

- o Table 4.3-1 Item 13;
- o Table 3.3-2 Items 4c, 4e, 5c and 7b; and
- o Table 4.3-2 Items 1c, 4d, 4e and 5c.

Footnote * on page 3/4 3-43 was deleted since no references to it on the subject page currently exist.

These changes do not affect any of the current Unit 1 requirements although they are equally applicable to Unit 1.

Justification for the Change

Several Reactor Trip System Instrumentation and Engineered Safety Features Actuation System Instrumentation Functional Units (Items)/Monitors are also included in the Accident Monitoring Instrumentation Specification (3/4.3.3.6). Footnotes have previously been incorporated to indicate which of these instruments/monitors also appear in Specification 3/4.3.3.6. However, these cross references were inconsistent in that some were omitted either from the text of the subject tables or the footnote itself at the bottom of the page. These changes provide a consistent reference to remind the operator that these subject monitors are also included in the Accident Monitoring Instrumentation Specification.

Pages affected.
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Page 3/4 3-21
Page 3/4 3-23
Page 3/4 3-36
Page 3/4 3-39
Page 3/4 3-40
Page 3/4 3-43

5. **Description of the Change**

Add "(Common System)" after the name of functional unit 11 in Table 3.3-2, 3.3-3, and 4.3-2.

Add footnote * to Table 3.3-6 stating "This instrumentation is common to Units 1 and 2".

Add "(Common System)" after METEOROLOGICAL INSTRUMENTATION for 3/4.3.3.4 and after FUEL HANDLING BUILDING POST ACCIDENT VENTILATION SYSTEM for 3/4.9.12, and after Electric Steam Boiler Isolation on Table 3.3-11.

This change does not affect any of the current Unit 1 requirements.

Justification for the Change

Those systems that are shared by Units 1 and 2 are identified as common systems. As discussed in FSAR Subsection 2.3.3 only one meterological tower is used for the site and as discussed in FSAR Subsection 9.4.2 the Fuel Handling Building Post Accident Ventilation System is common to Units 1 and 2. The identification of systems as common to both units is necessary to show those specifications that are not applicable on a per unit basis.

Pages affected.
- Page 3/4 3-24
- Page 3/4 3-34
- Page 3/4 3-44
- Page 3/4 3-53
- Page 3/4 3-54
- Page 3/4 3-80
- Page 3/4 9-14

6. <u>Description of the Change</u>

Change the word "the" to "either" in table notation i of Table 3.3-2 and in Specification 3.9.12.

This change does not affect any of the current Unit 1 requirements.

<u>Justification for the Change</u>

This change applies to the Fuel Handling Building Post Accident Ventilation Actuation System which is a common system. Because it is a common system the specification will apply if either unit is in one of the applicable MODES.

Pages affected.
Page 3/4 3-25
Page 3/4 9-14

7. **Description of the Change**

Add a footnote which delays the applicability of Specification 3/4.3.3.1 and 3/4.3.3.6 to the Unit 2 Containment Area High Range Radiation Monitors (RE-0005 and RE-0006) until the initial entry of Unit 2 into Mode 2.

This change does not affect any of the current Unit 1 requirements.

Justification for the Change

This exception was approved for Unit 1 as documented in GPC letter SL-2016, dated February 17, 1987. Due to the absence of fission products prior to the initial entry into MODE 2, and hence the lack of potential radiological consequences of an accident, the Post Accident Monitoring function of the Containment Area High Range Radiation Monitors (RE-0005 and RE-0006) is not necessary prior to initial criticality.

Pages affected.
- Page 3/4 3-46, Table 3.3-4
- Page 3/4 3-48, Table 4.3-3
- Page 3/4 3-58

8. **Description of the Change**

Change "The plant" to "Unit 1" in Specification 4.3.3.2 and move the wording of the footnote to the bases.

This change does not affect any of the current Unit 1 requirements.

Justification for the Change

As described in FSAR Section 3.7.4.1 only one complete set of seismic instrumentation is provided for the site and meets the recommendations of Reg. Guide 1.12, Rev. 1, April, 1974. This instrumentation has been operational since Unit 1 was licensed and has been operated in accordance with the Unit 1 Technical Specifications. The changes are made to indicate that only the one set of instrumentation exists. The reference to "the plant" entering MODE 5 or below for surveillance of instrumentation not accessible is revised to "Unit 1" to avoid confusion since the instrumentation is physically located in Unit 1.

Pages affected.
Page 3/4 3-50
Page B 3/4 3-4

9. <u>Description of the Change</u>

Add "Unit 1" to items 2 a, b, c, item 3a and items 5a and b of Table 3.3-5 and 4.3-4.

This change does not affect any current Unit 1 requirement.

<u>Justification for the Change</u>

This change clearly identifies the location of the instrumentation which is only installed in Unit 1.

Pages affected.
Page 3/4 3-51
Page 3/4 3-52

10. Description of the Change

Revise Specification 3.3.3.7 to show a 5 ppm setpoint instead of 2 ppm for alarm/trip setpoint for the Chlorine Detection Systems. This change is currently under review by the NRC for Unit 1.

The instrument designator was revised from "A" to "loop". This change does not affect the current Unit 1 requirement.

Justification for the Change

This change is being made to be consistent with a Technical Specification change that is currently under review for the Vogtle Unit 1 Technical Specifications and is expected to be approved by the NRC prior to the issuance of the Unit 2 license. The request for the change to the Unit 1 Technical Specifications was contained in Georgia Power letter SL-2603 of August 26, 1987. The evaluation and justification contained in that letter is applicable to Unit 2 as well as to Unit 1.

The instrument designator was revised from "A" to "loop" to avoid the confusion with the "A" used to indicate items that are common to both units. "A" originally represented "analyzers." This revision of the instrument designator does not change any of the requirements for Unit 1.

Pages affected.
Page 3/4 3-63

11. Description of the Change

Delete item 1.d from Table 4.3-5 and 3.3-9 "Control Building (Floor Drains) Sumps Effluent Line (RE-17646)."

This change is consistent with a change that will be requested for the Unit 1 Technical Specifications.

Justification for the Change

The basis for this change is a piping modification which eliminates the need for Monitor RE-17646. The modification involves the rerouting of the control building sump discharge piping to a flow path monitored by existing Monitor RE-0848. Monitor RE-0848 has range and sensitivity similar to those of RE-17646 and initiates automatic termination of a radioactive release. This change will be requested by GPC letter SL-3777 which is expected to be submitted by January 31, 1988. The change applies equally to Unit 2.

Pages affected.
Page 3/4 3-66
Page 3/4 3-68

12. Description of the Change

Revise Table 3.3-11 to show that the Electric Steam Boiler Isolation instrument channels are common by adding an "A" to the instrument numbers. Add the Unit 2 Steam Generator Blowdo.n Line Isolation and Letdown Line Isolation instruments with their Unit 2 locations.

This change does not affect any of the current Unit 1 requirements.

Justification for tue Change

These changes provide the Unit 2 room locations for the Unit 2 instruments and the indication of which instruments are common for both units.

Pages affected.
Page 3/4 3-80

13. Description of the Change

Add a footnote which delays the initial applicability of Specification 3/4.4.4 to Unit 2 until its initial entry into Mode 2.

This change does not affect any of the current Unit 1 requirements.

Justification for the Change

This exception was approved for Unit 1 and was included in the Unit 1 low power Technical Specifications (Appendix A to Operating License NPF 61). This particular component has a history of problems and retests are a likely requirement during the initial test program. This exception eliminates the necessary to heat up the Reactor Coolant System using Reactor Coolant Pumps for the specific purpose of completing this test. This would subject the plant to unnecessary thermal cycles. The lack of fission products prior to initial criticality is the basis for this exception.

Pages affected.
Page 3/4 4-10

14. <u>Description of the Change</u>

Revise action a of Specification 3.6.1.6 to indicate how the required report of inadequate tendon lift-off force should address both units even though the measurements can only be made on Unit 1. In general revise the specification to reflect the fact that Unit 2 does not have detensionable tendons.

This change does not affect the current Unit 1 requirements.

<u>Justification for the Change</u>

Specification 3/4.6.1.6 has been revised to reflect Unit 2. As described in FSAR Section 3.8.1.7.2, Unit 2 containment does not have detensionable tendons. Therefore, consistent with the recommendations of revision 2 of Regulatory Guide 1.35, Unit 2 is subject to visual inspection of end anchorages, adjacent concrete surfaces, and containment surfaces during Type A tests, and a visual examination of the containment tendon anchorages at the 1, 3, 5, and 10 year, etc. intervals.

Action a is only directly applicable to Unit 1 because it is associated with lift-off forces and Unit 2 is not required to measure lift-off forces. However, if Unit 1 does not meet the lift-off force requirements, an engineering evaluation of the applicability of the nonconformance to Unit 2 should be performed. The determination of applicability, and action for Unit 2 should be based on the results of the Unit 1 evaluation. Therefore, the Unit 2 reporting time is keyed to the Unit 1 final evaluation. The Unit 2 containment tendons were initially stressed over 1 year after the Unit 1 tendons. It is anticipated that any required evaluation of tendon lift-off forces not within predicted limits would lead to an isolated case of instrument error, uneven lift-off of the end anchorage or a reassessment of actual losses. If the evaluation of Unit 1 leads to a generic concern of unpredicted degradation, the allowed evaluation time frame for Unit 2 is small compared to the more than 1 year difference in the initial stressing of the tendons.

The new ACTION b was written because surveillance 4.6.1.6.1.1 items b and e while not associated with lift-off forces are only applicable to Unit 1 and therefore, should also be evaluated for Unit 2 in a manner similar to the rewritten ACTION a.

ACTION c was written to be applicable to 4.6.1.6.1.2, 4.6.1.6.2 and 4.6.1.6.3 where the equivalent surveillance is done on both units.

Pages affected.
- Page 3/4 6-8
- Page 3/4 6-10
- Page B 3/4 6-2

15. <u>Description of the Change</u>

Change the percentage of instrument span indicated for the lower and upper spray additive tank volume limits from 87.7% and 97.4% to 89.9% and 97.2% including associated Bases.

This change will be requested for Unit 1.

Provide Unit 2 specific values for the eductor suction flow test requirements of Specification 4.6.2.2.d.

This change does not affect the current Unit 1 requirements.

<u>Justification for the Change</u>

The change to the tank volumes makes the percentages of level span consistent with the volumes specified in gallons. The currently specified percentages ensure that loss-of-coolant accident consequences would be within FSAR limits, however the amount of margin would be less than originally intended. The change increases the margin to the intended level. The change will be requested for Unit 1 in GPC letter SL-3777 which is expected to be submitted by January 31, 1988, and is applicable to Unit 2.

The eductor suction flow rate will be determined for Unit 2 during the Unit 2 preoperational tests and may differ slightly from the Unit 1 values. Therefore, a space has been added for incorporation of those values when they have been determined.

Pages affected.
Page 3/4 6-14
Page B 3/4 6-3

16. <u>Description of the Change</u>

Add a footnote which delays the initial applicability of Specification 3/4.7.1.2 to Unit 2 until its initial entry into MODE 2.

This change does not affect any of the current Unit 1 requirements.

Change 175 gpm to 150 gpm as the value of the test flow in 4.7.1.2.1.a.1).

This change is consistent with a change that will be requested for the Unit 1 Technical Specifications.

<u>Justification for the Change</u>

This exception was approved for Unit 1 and was included in the Unit 1 low power Technical Specifications (Appendix A to Operating License NPF 61). Experience with Unit 1 has demonstrated that a number of adjustments to this system may have to be made during preoperational testing. Therefore in order to preclude multiple heatups and pressurizations on the primary and secondary systems, the proposed footnote was added to allow these adjustments to be made prior to initial entry into MODE 2. Delaying the applicability of this Specification until the initial entry into MODE 2 is acceptable due to the lack of decay heat. Without this exception it may be necessary to heatup the Reactor Coolant System using Reactor Coolant Pumps for the specific purpose of making these adjustments. This would subject the plant to unnecessary thermal cycles.

This change provides a more appropriate flow requirement for surveillance testing of the motor driven AFW pumps. The presently specified 175 gpm is the sizing basis for the recirculation lines and is based on the manufacturers original recommendation for pump protection. The pump manufacturer has stated that 150 gpm is equally acceptable to protect the pumps during recirculation. The change to 150 gpm provides margin to account for design and construction tolerances in the recirculation lines. Since pump head is constant in the 150-175 gpm flow region, 1605 psig is still the appropriate discharge pressure. This change will be requested for Unit 1 in GPC letter SL-3831 which is expected to be submitted by January 31, 1988. The change is equally applicable to Unit 2.

Pages affected.
Page 3/4 7-4

17. Description of the Change

Add a footnote which delays the applicability of Specification 3/4.7.7 to the Unit 2 Piping Penetration Area Filtration and Exhaust System until the initial entry of Unit 2 into Mode 2.

Add a place for the Unit 2 flow rate in 4.7.7.b, d, e, f.

Remove the "C" from the instrument numbers given in 4.7.7.d.

This change does not affect any current Unit 1 requirement.

Justification for the Change

This exception was approved for Unit 1 as documented in GPC letters SL-1952, dated February 4, 1987, and SL-2016, dated February 17, 1987. Flow balancing of HVAC systems is an iterative, time consuming process requiring many adjustments be made prior to completion of the testing. Due to the absence of fission products prior to the initial entry into MODE 2, operability of the Piping Penetration Area Filtration and Exhaust System on Unit 2 is not necessary until initial criticality.

The piping penetration area filtration and exhaust system flow rate for Unit 2 may differ from that of Unit 1 but will not be determined until later in the test program.

The instrument numbers PDIC-2550 and PDIC-2551 are being corrected to read PDI-2550 and PDI-2551.

Pages affected.
Page 3/4 7-17
Page 3/4 7-18

18. Description of the Change

The change deletes the requirement for a mid-first cycle unit shutdown to inspect 100% of the snubbers.

This change does not affect current Unit 1 requirements.

Justification for the Change

These inspections were performed on Unit 1. The results showed no significant snubber degradation. The snubbers provided for Unit 2 are of the same design, are subject to the same installation and preoperational requirements and will be subjected to similar operating conditions. In light of the success of the snubber inspections program for Unit 1 it is proposed to conduct the initial 100% inspection at the first outage and to avoid a costly mid-cycle shutdown for snubber inspection.

Pages affected.
Page 3/4 7-19

19. <u>Description of the Change</u>

The component numbers given in Specifications 4.7.13, 3.8.2.1, 3.8.3.1, and Table 3.8-1 are revised to show both units.

This change does not affect the current Unit 1 requirements.

<u>Justification for the Change</u>

The component numbers currently start with the number "1" which indicates Unit 1. The change replaces it with a "1/2" indicating components exist for each unit.

Pages affected.
- Page 3/4 7-31
- Page 3/4 8-11
- Page 3/4 8-15
- Page 3/4 8-16
- Page 3/4 8-22
- Page 3/4 8-23
- Page 3/4 8-24

20. **Description of the Change**

Add a footnote which exempts the Unit 2 refueling canal from boron concentration requirements during initial fuel load, provided that the Unit 2 refueling canal level is verified to be below the reactor vessel flange elevation at least once every 12 hours.

This change does not affect any current Unit 1 requirement.

Justification for the Change

This exception was approved for Unit 1 and was included in the Unit 1 low power Technical Specifications (Appendix A to Operating License NPF-61). During initial fuel load it is not necessary to completely flood the refueling canal. It is only necessary to provide a small amount of water to lubricate the fuel transfer equipment. When the refueling canal is filled to the limited degree necessary, the water in the canal does not communicate with the water in the reactor vessel. Since reactivity control is not a concern for movement of new fuel through the transfer tube, boration of the water in the canal is not necessary.

Pages affected.
Page 3/4 9-1

21. <u>Description of the Change</u>

Change "the" to "each" in 4.11.4.2.

This change does not affect any Unit 1 requirements.

<u>Justification for the Change</u>

The change shows that dose contributions are to be determined from each unit.

Pages affected.
Page 3/4 11-18

22. <u>Description of the Change</u>

Add new sections 5.6.1.3 and 5.6.1.4 to reflect the spent fuel storage racks to be installed in the West Spent Fuel Pool.

This change does not affect the current Unit 1 requirements.

<u>Justification for the Change</u>

As discussed in letter GN-1422 of December 23, 1987, the West Spent Fuel Pool will be licensed to contain 2098 fuel assemblies. The storage racks are designed to be free-standing and to perform their function whether standing alone or adjacent to other free-standing spent fuel racks, and whether or not they contain spent fuel. It is expected that eight of the 20 racks will be installed when the Unit 2 license is issued and that approximately 12 more will be added after Unit 2 initial fuel loading.

Pages affected.
Page 5-5

23. <u>Description of the Change</u>

Replace 6.2.2.b with a revised 6.2.2.b (including a footnote) that addresses two unit operation, and revise Table 6.2-1 to apply to two unit operation, also change "the" to "either" in 6.2.2.c.

This does not affect the current Unit 1 requirements.

<u>Justification for the Change</u>

With two units in operation and one control room, the minimum requirements per shift for onsite staffing of Nuclear Power Units must comply with the requirements of 10 CFR 50.54(m)(2). These changes meet that requirement. The revision to Table 6.2-1 is based on NUREG-0452 draft Rev. 5 (the draft issued to GPC by the NRC as a basis for the initial markup of the Unit 1 Technical Specifications) for two units with a common control room.

The change from "the" to "either" in 6.2.2.c represents the fact that these Technical Specifications will be applicable to both units.

Pages affected.
Page 6-1
Page 6-5

24. Description of the Change

On Figure 6.2-1 and page 6-6 change the title of Manager-Nuclear Performance and Analysis to Manager-Nuclear Performance and Radiological Safety.

On Figure 6.2-2 add the new position of Outages and Planning Manager reporting to the Plant Support Manager. Delete the Unit 1 Planning, Scheduling, and Work Control Methods Superintendent. Change the title for the Maintenance Superintendent III position to Manager Maintenance. Add the new position of Manager Health Physics and Chemistry, who has authority over the Health Physics Superintendent, the Chemistry Superintendent, and the new position of Technical Support Health Physics/Chemistry Superintendent.

This change is the same as what has been requested for Unit 1.

Justification for the Change

This change was requested by letter SL-3874 of December 28, 1987. The new position of Outages and Planning Manager provides a higher level of management oversight to work control and outage planning. The position of Unit 1 Planning, Scheduling and Work Control Methods Superintendent still exists and reports to the Outages and Planning Manager. The shifting of the outages and planning functional unit from the Plant Manager's to the Plant Support Manager's responsibility places it in a more appropriate organization and allows the Plant Manager to focus on and devote more attention to operating activities. The change in title from Maintenance Superintendent III to Manager - Maintenance is an upgrade of the position and does not affect the duties and responsibilities of the position. The new position of Technical Support Health Physics/Chemistry Superintendent is added to assume responsibility for support functions previously performed by the Health Physics and Chemistry groups. This change allows the Health Physics and Chemistry personnel to devote their full attention to plant operating activities. Addition of the Manager-Health Physics and Chemistry provides a higher level of management oversight and ensures full coordination and communication between subordinate groups.

Pages affected.
- Page 6-3
- Page 6-4
- Page 6-6

25. <u>Description of the Change</u>

Change "this operating license" to "these operating licenses" in item 6.4.2.7.d.

This change does not affect any current Unit 1 requirements.

<u>Justification for the Change</u>

This change reflects the fact that Unit 1 and Unit 2 will each have an operating license.

Pages affected.
Page 6-10

26. Description of the Change

Change "the unit" to "the affected unit" in Specification 6.6.1.d.

This change does not affect any current Unit 1 requirements.

Justification for the Change

The change reflects the fact that there are two units and that exceeding a safety limit on one unit does not require the other unit to shut down.

Pages affected.
Page 6-13

27. <u>Description of the Change</u>

Change "the unit" to "either unit" in Specification 6.8.1.1.

This change does not affect the current Unit 1 requirements.

<u>Justification for the Change</u>

This change is intended to clarify the application of this requirement to either unit.

Pages affected.
Page 6-16

28. Description of the Change

Add footnotes for 6.8.1.2 and 6.8.1.3 to indicate that a single submittal may be made for both units, and change "unit" to "plant" in 6.8.1.3.

This does not affect the current Unit 1 requirements.

Justification for the Change

This change maintains the same reporting requirement but adds a footnote to the effect that single submittals may be made for Units 1 and 2.

These changes meet the intent of NUREG-0452 draft Rev. 5 (the draft issued to GPC by the NRC for a basis for the initial markup of the Unit 1 Technical Specifications).

Pages affected.
Page 6-17

29. <u>Description of the Change</u>

Change "the unit or station" to "each unit" in the second paragraph of page 6-19.

This change does not affect the current Unit 1 requirements.

<u>Justification for the Change</u>

The change shows that dose contributions are to be determined from each unit.

Pages affected.
Page 6-19

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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 Technical 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 flux signals (normalized to their full power sum and expressed as their percentage) 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.

CONTAINMENT INTEGRITY

1.7 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.

b. All equipment hatches are closed and sealed,

c. Each air lock is 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

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

CORE ALTERATIONS

1.9 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.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration in MeV, for the isotopes with half lives greater than 14 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 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.

FREQUENCY NOTATION

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

GASEOUS WASTE PROCESSING SYSTEM

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

IDENTIFIED LEAKAGE

1.15 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 Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.16 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.

MEMBER(S) OF THE PUBLIC

1.17 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.

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

OPERABLE - OPERABILITY

1.19 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, electrical 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).

OPERATIONAL MODE - MODE

1.20 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 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (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.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.23 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.

PURGE - PURGING

1.24 PURGE or PURGING shall be any 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

1.25 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

1.26 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

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

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Sections 50.72 and 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all 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

1.30 The SITE BOUNDARY shall be the exclusion boundary line as shown in Figure 5.1-1.

SLAVE RELAY TEST

1.31 A SLAVE RELAY TEST shall be the energization of each slave relay and 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 wet wastes 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 when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.34 A STAGGERED TEST BASIS shall consist of:

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.

THERMAL POWER

1.35 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.36 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

1.37 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.38 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.

VENTILATION EXHAUST TREATMENT SYSTEM

1.39 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 Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.40 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.

TABLE 1.1

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.
N.A.	Not applicable.
P	Completed prior to each release.

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 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**	$\leq$ 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.

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 (NI-0041, NI-0042, NI-0043, NI-0044), pressurizer pressure (PI-0455A, B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), and the highest operating loop coolant temperature (T_{avg}) (TI-0412, TI-0422, TI-0432, TI-0442, 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 6.6.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure (PI-0408, PI-0418, PI-0428, PI-0438) 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 6.6.1.

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 6.6.1.

and apply to each unit unless specifically noted

*Where specific instrument numbers are provided in parentheses, they are for information only, (to assist in identifying associated instrument channels or loops) and are not intended to limit the requirements to the specific instruments associated with the number.

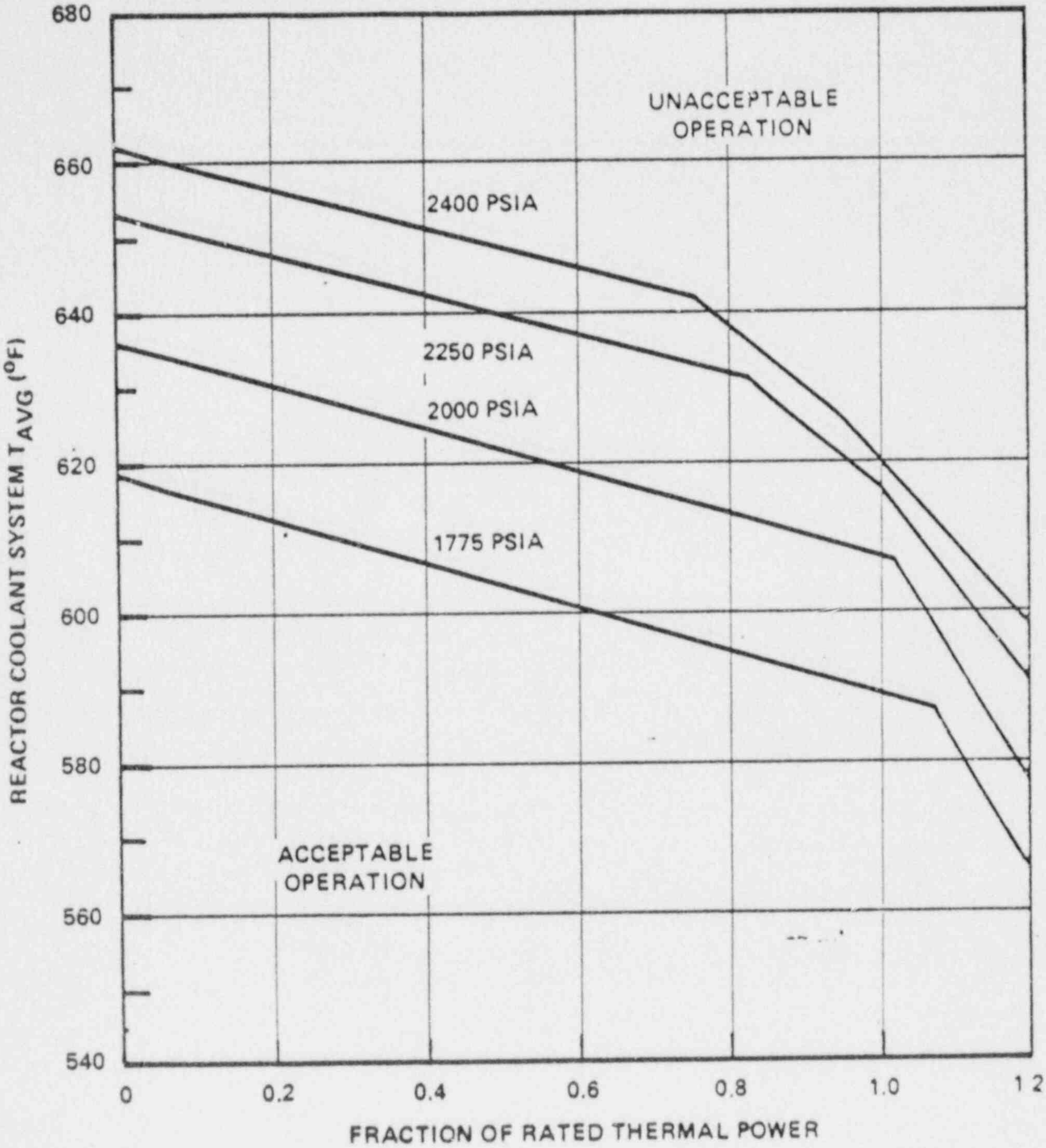

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT

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:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. 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 Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = The value from Column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)					
a. High Setpoint	7.5	4.56	0	$\leq$109% of RTP#	$\leq$111.3% of RTP#
b. Low Setpoint	8.3	4.56	0	$\leq$25% of RTP#	$\leq$27.3% of RTP#
3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	1.6	0.50	0	$\leq$5% of RTP# with a time constant $\geq$2 seconds	$\leq$6.3% of RTP# with a time constant $\geq$2 seconds
4. Power Range, Neutron Flux, High Negative Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	1.6	0.50	0	$\leq$5% of RTP# with a time constant $\geq$2 seconds	$\leq$6.3% of RTP# with a time constant $\geq$2 seconds
5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B)	17.0	8.41	0	$\leq$25% of RTP#	$\leq$31.1% of RTP#
6. Source Range, Neutron Flux (NI-0031B, NI-0032B)	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C)	6.6	3.37	1.95 + 0.50	See Note 1	See Note 2
8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B)	4.9	1.54	1.95	See Note 3	See Note 4

#RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≥1960 psig**	≥1950 psig
10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	3.1	0.71	1.67	≤2385 psig	≤2395 psig
11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461)	8.0	2.18	1.67	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low (LOOP1 LOOP2 LOOP3 LOOP4 FI-0414 FI-0424 FI-0434 FI-0444 FI-0415 FI-0425 FI-0435 FI-0445 FI-0416 FI-0426 FI-0436 FI-0446)	2.5	1.87	0.60	≥90% of loop design flow*	≥89.4% of loop design flow*
13. Steam Generator Water Level Low-Low (LOOP1 LOOP2 LOOP3 LOOP4 LI-0517 LI-0527 LI-0537 LI-0547 LI-0518 LI-0528 LI-0538 LI-0548 LI-0519 LI-0529 LI-0539 LI-0549 LI-0551 LI-0552 LI-0553 LI-0554)	18.5	17.18	1.67	≥18.5%*** of narrow range instrument span	≥17.8% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	6.0	0.58	0	≥9600 volts (70% bus voltage)	≥9481 volts (69% bus voltage)
15. Underfrequency - Reactor Coolant Pumps	3.3	0.50	0	≥57.3 Hz	≥57.1 Hz

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. Channel Calibration shall ensure that these time constants are adjusted to these values.

***Until resolution of the Veritrak transmitter uncertainty issue this setpoint will be set at ≥22.5% (Unit 1) and ≥23.5% (Unit 2) of narrow range instrument span.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
16. Turbine Trip					
a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163)	N.A.	N.A.	N.A.	≥580 psig	≥500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥96.7% open	≥96.7% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6 (NI-0035B, NI-0036B)	N.A.	N.A.	N.A.	≥1 x 10^{-10} amp	≥6 x 10^{-11} amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤10% of RTP#	≤12.3% of RTP#
2) P-13 input (PI-0505, PI-0506)	N.A.	N.A.	N.A.	<10% RTP# Turbine Impulse Pressure Equivalent	<12.3% RTP# Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤48% of RTP#	≤50.3% of RTP#

#RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. Power Range Neutron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A.	≤50% of RTP#	≤52.3% of RTP#
e. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	N.A.	N.A	≥10% of RTP#	≥7.7% of RTP#
f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	N.A.	N.A.	N.A.	≤10% RTP# Turbine Impulse Pressure Equivalent	≤12.3% RTP# Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

#RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_1	$\leq$	1.10;
K_2	=	0.012/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	588.5°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.00056/psig;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform variable, s^{-1};

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

(1) For $q_t - q_b$ between -33.5% and + 6.5%, $f_1(\Delta I) = 0$, 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;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 33.5%, the ΔT Trip Setpoint shall be automatically reduced by 1.27% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 6.5%, the ΔT Trip Setpoint shall be automatically reduced by 0.83% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.5%.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{(1 + \tau_3 S)}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{(1 + \tau_7 S)}\right) \left(\frac{1}{(1 + \tau_6 S)}\right) T - K_6 \left[T \left(\frac{1}{(1 + \tau_6 S)}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD manifold instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-leg compensator for ΔT, $\tau_1 \geq 8$ S, $\tau_2 \leq 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, τ_3=0s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_4	$\leq$	1.089,
K_5	$\geq$	0.02/°F for increasing average temperature and $\geq$ 0 for decreasing average temperature,
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation,
τ_7	=	Time constants utilized in the rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;
K_6	$\geq$	0.0013/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,
T	=	Average Temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.5°F),
S	=	Laplace transform variable, s^{-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 2.4% of ΔT span.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 2.0, but in accordance with 10 CFR 50.36 are not 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.

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 (R Grid) correlation. The W-3 (R Grid) DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform 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% probability at a 95% 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 reactor core safety limits which are determined for a range of reactor operating conditions. The core limits represent the loci of points of THERMAL POWER, REACTOR COOLANT SYSTEM pressure and average temperature which satisfy the following criteria:

A. The average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid (far left line segment in each curve).

B. The minimum DNBR is not less than the design limit (all the other line segments in each curve).

C. The hot channel exit quality is not greater than the upper limit of the quality range of the W-3 (R-Grid) correlation which is 15% (middle line segment on Reactor Coolant System pressure curves, 2400 psia and 2250 psia; this is not a limiting criterion for this plant).

REACTOR CORE (Continued)

These 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:

$$F^N_{\Delta H} = 1.55\ [1+ 0.3\ (1-P)]$$

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 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips 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 (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

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 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 Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor 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.

Manual Reactor Trip

The Reactor 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 Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% 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 protection for control rod drop accidents. At high power a single or multiple 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 DNBRs will be greater than 1.30.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide 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% of RATED THERMAL POWER unless manually blocked when P-10 becomes active. Although no explicit credit was taken for the source range instrumentation in the accident analyses, operability requirements in the Technical Specifications will ensure that the source range instrumentation is available to mitigate the consequences of an inadvertent control bank withdrawal in MODES 3, 4, and 5.

Overtemperature ΔT

The Overtemperature ΔT 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.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of 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 Releases."

Pressurizer Pressure

In each of the pressurizer 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% of RATED THERMAL POWER or turbine impulse chamber pressure at approximately 10% 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 approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

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% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% 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 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

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.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide 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 loss of power to the reactor coolant pump bus circuit breakers shall not exceed 1.2 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 second.

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% 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 Reactor trip from the Turbine trip is automatically blocked by P-9 (a power level of approximately 50% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-9.

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 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 trip (i.e., prevents premature block of Source Range trip), and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

Reactor Trip System Interlocks (Continued)

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

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the above trips.

P-9 On increasing power, P-9 automatically enables Reactor trip on Turbine trip. On decreasing power, P-9 automatically blocks Reactor trip on Turbine trip.

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

P-13 Provides input to P-7.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION 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 1 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 of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 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 requirements. Exceptions to these requirements are stated in the individual specifications.

(and apply to each unit unless specifically noted)

*Where specific instrument numbers are provided in parenthes~~i~~es, they are for information only (to assist in identifying associated instrument channels or loops) and ~~are~~ not intended to limit the requirements to the specific instruments associated with the number.

**For the purposes of determination of TOTAL NO. OF CHANNELS, CHANNELS TO TRIP, MINIMUM CHANNELS OPERABLE requirements, APPLICABLE MODES, and ACTION requirements, the number of channels provided is on a per-unit basis unless specifically noted.

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 three 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 has 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 Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i);

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities;

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements; and

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% $\Delta k/k$.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the SHUTDOWN MARGIN less than 1.3% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 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 1.3% $\Delta k/k$:

a. Within 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 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 MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. With K_{eff} less than 1, 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; and

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

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

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% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1d, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

SHUTDOWN MARGIN - MODES 3, 4 AND 5

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limits shown in Figures 3.1-1 and 3.1-2.

APPLICABILITY: MODES 3, 4 AND 5.

ACTION:

With the SHUTDOWN MARGIN less than the required value, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 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 the required value:

a. Within an 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(s) 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.

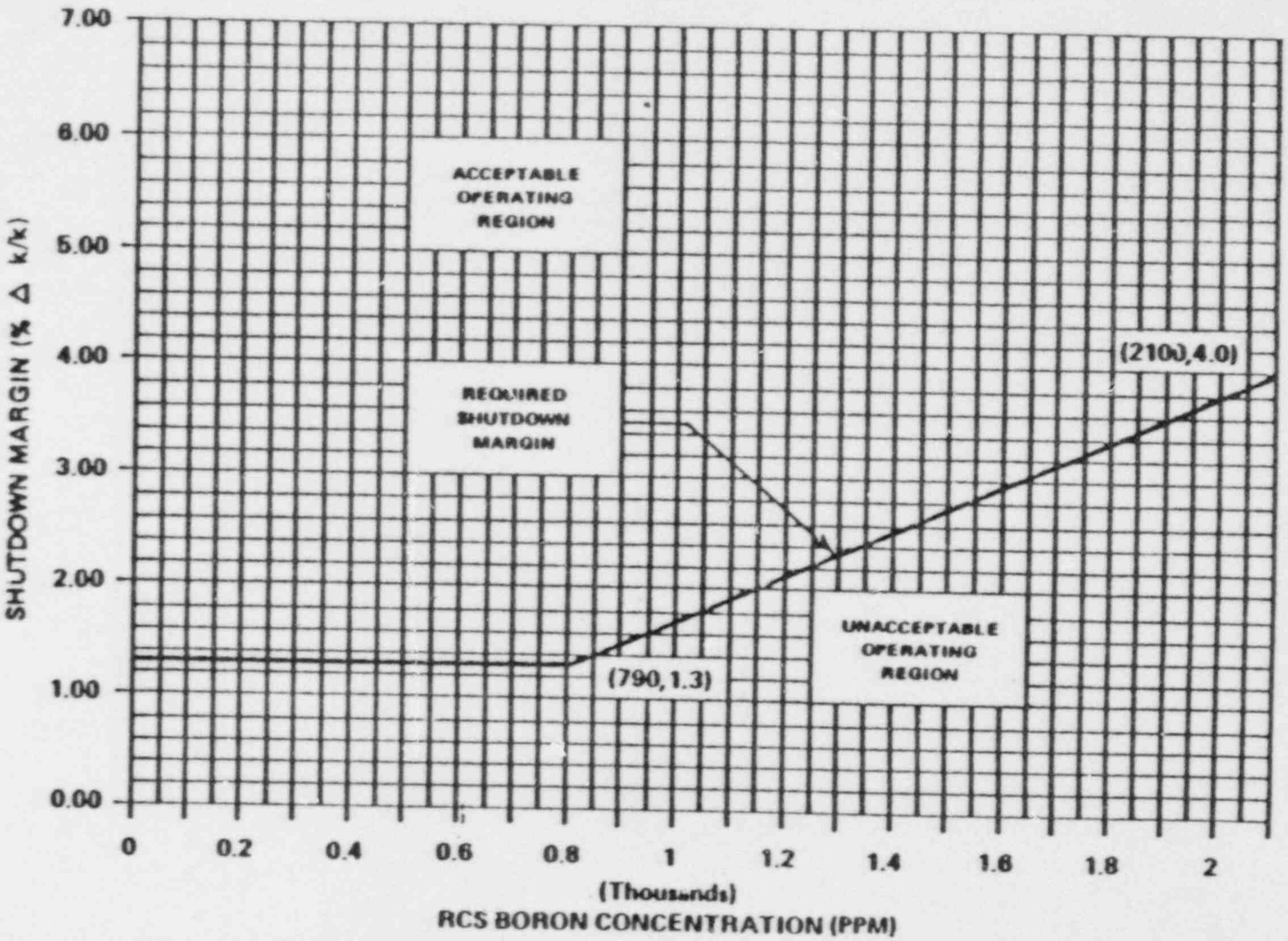

FIGURE 3.1-1 REQUIRED SHUTDOWN MARGIN FOR MODES 3 AND 4 (MODE 4 WITH AT LEAST ONE REACTOR COOLANT PUMP RUNNING)

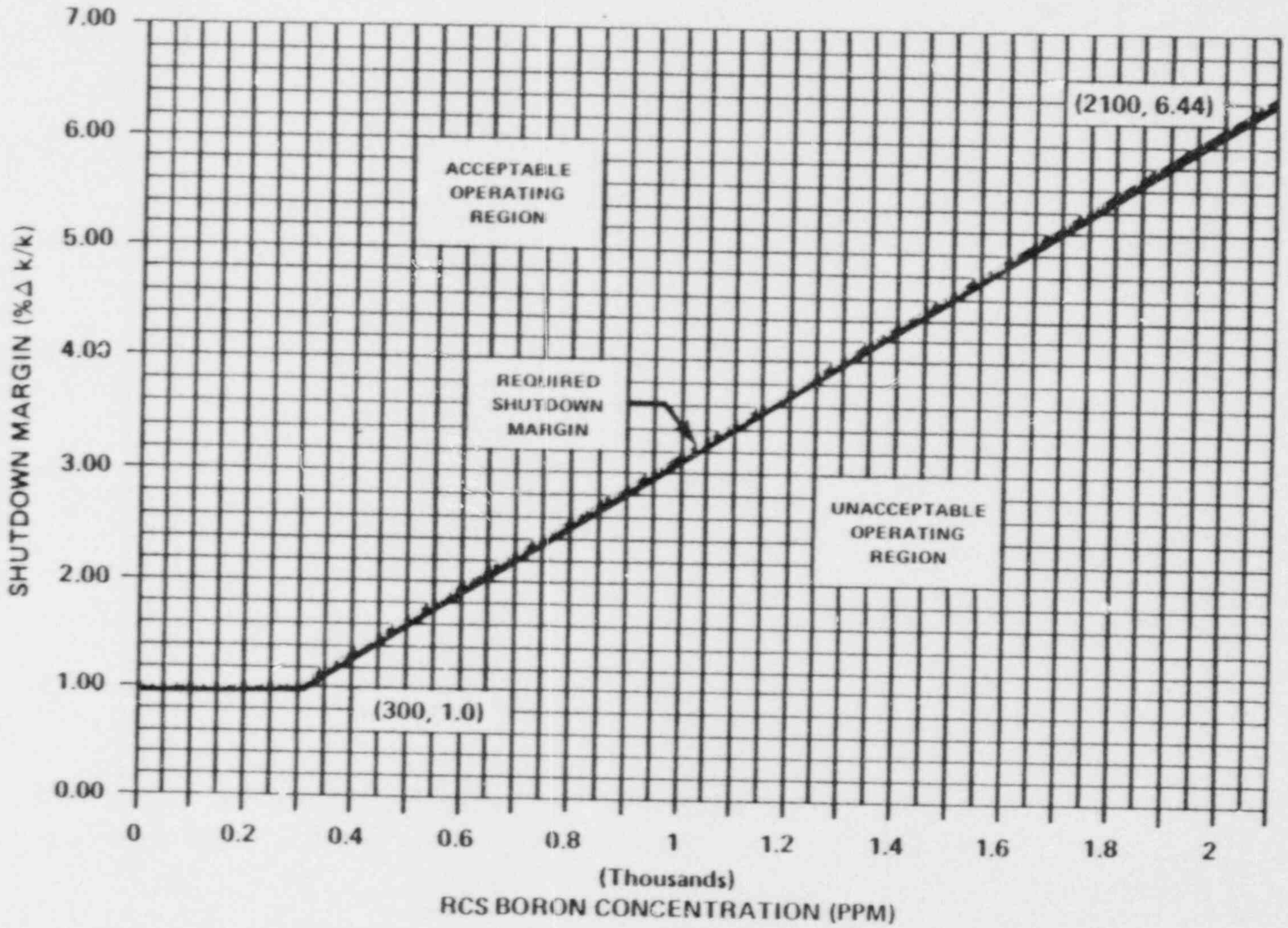

FIGURE 3.1-2 REQUIRED SHUTDOWN MARGIN FOR MODE 5 (MODE 4 WITH NO RCPs RUNNING)

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

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

b. Less negative than - 4.0 x 10^{-4} $\Delta k/k/^{\circ}F$ for the 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 $\Delta k/k/^{\circ}F$ 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. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.8.2, 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.

*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 -3.1×10^{-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 -3.1×10^{-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 (TI-0412, TI-0422, TI-0432, TI-0442) (T_{avg}) shall be greater than or equal to 551°F.

APPLICABILITY: MODES 1 and 2* **.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°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 551°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} (TI-0412, TI-0422, TI-0432, TI-0442) is less than 561°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.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

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

b. A 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, 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 when the boric acid storage tank is a required water source, verify that the applicable portions of the auxiliary building (TISL 12410 or TISL 12411, TISL 12412 or TISL 12413, TISL 12414 or 12415, TISL 12416 or TISL 12417, TISL 20900 or TISL 20901, TISL 20902 or TISL 20903, and TISL 20904 or TISL 20905) are ≥72°F the portions of the flow path for which ambient temperature indication are not provided are ≥ 65°F, 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.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following boron injection flow paths shall be OPERABLE:

a. One or more flow paths from the boric acid storage tank via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and/or

b. One or more flow paths from the refueling water storage tank via charging pumps to the RCS.

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

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as required by Figure 3.1-2 at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days when the boric acid tank is a required water source, verify that the applicable portions of the auxiliary building (TISL 12410 or TISL 12411, TISL 12412 or TISL 12413, TISL 12414 or 12415, TISL 12416 or TISL 12417, TISL 20900 or TISL 20901, TISL 20902 or TISL 20903, and TISL 20904 or TISL 20905) are $\geq$ 72°F and the portions of the flow path for which ambient temperature indication are not provided are $\geq$65°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 the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the 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.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least once per 18 months the above required charging pump shall be demonstrated OPERABLE by verifying that the flow path required by Specification 3.1.2.1a is capable of delivering at least 30 gpm to the RCS.

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 as required by Figure 3.1-2 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 At least two charging pumps shall be demonstrated OPERABLE by testing pursuant to Specification 4.1.2.2c.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 9504 gallons (19% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 70832 gallons (5% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2000 ppm and 2200 ppm, and

3) A minimum solution temperature of 54°F (TI-10982).

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) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is $<72°F$, verify the boric acid storage tank solution temperature is $\geq 65°F$.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when it is the source of borated water and the outside air temperature is less than 50°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 36674 gallons (81% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 631478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2000 ppm and 2100 ppm,

3) A minimum solution temperature of 54°F, and

4) A maximum solution temperature of 116°F (TI-10982).

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

ACTION:

a. With the Boric Acid Storage Tank inoperable and being used as one of the above required borated water sources, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN as required by Figure 3.1-2 at 200°F; restore the Boric Acid Storage Tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. 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

4.1.2.6 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) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is $\leq$ 72°F, verify the boric acid storage tank solution temperature is $\geq$ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when the outside air temperature is less than 50°F.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All 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 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 one rod trippable but 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 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 3.1-3. 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.

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 1 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 3.1-3. 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 its group step counter demand height 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 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 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

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE CONTROL OR SHUTDOWN ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Decrease in Reactor Coolant Inventory

Inadvertent Opening of a Pressurizer Safety or Relief Valve

Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundry That Penetrate Containment

Loss-of-Coolant-Accidents

Increase in Heat Removal by the Secondary System (Steam System Piping Rupture)

Spectrum of Rod Cluster Control Assembly Ejection Accidents.

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control and shutdown 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 nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating 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.

c. The provisions of Specification 3.0.4 are not applicable when verifying system operability following repair.

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.

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION 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 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.

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

**See Special Test Exceptions Specification 3.10.5.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual 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} (TI-0412, TI-0422, TI-0432, TI-0442) greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any 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 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.

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 1 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 Bank 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.

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 3.1-3.

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:

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

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

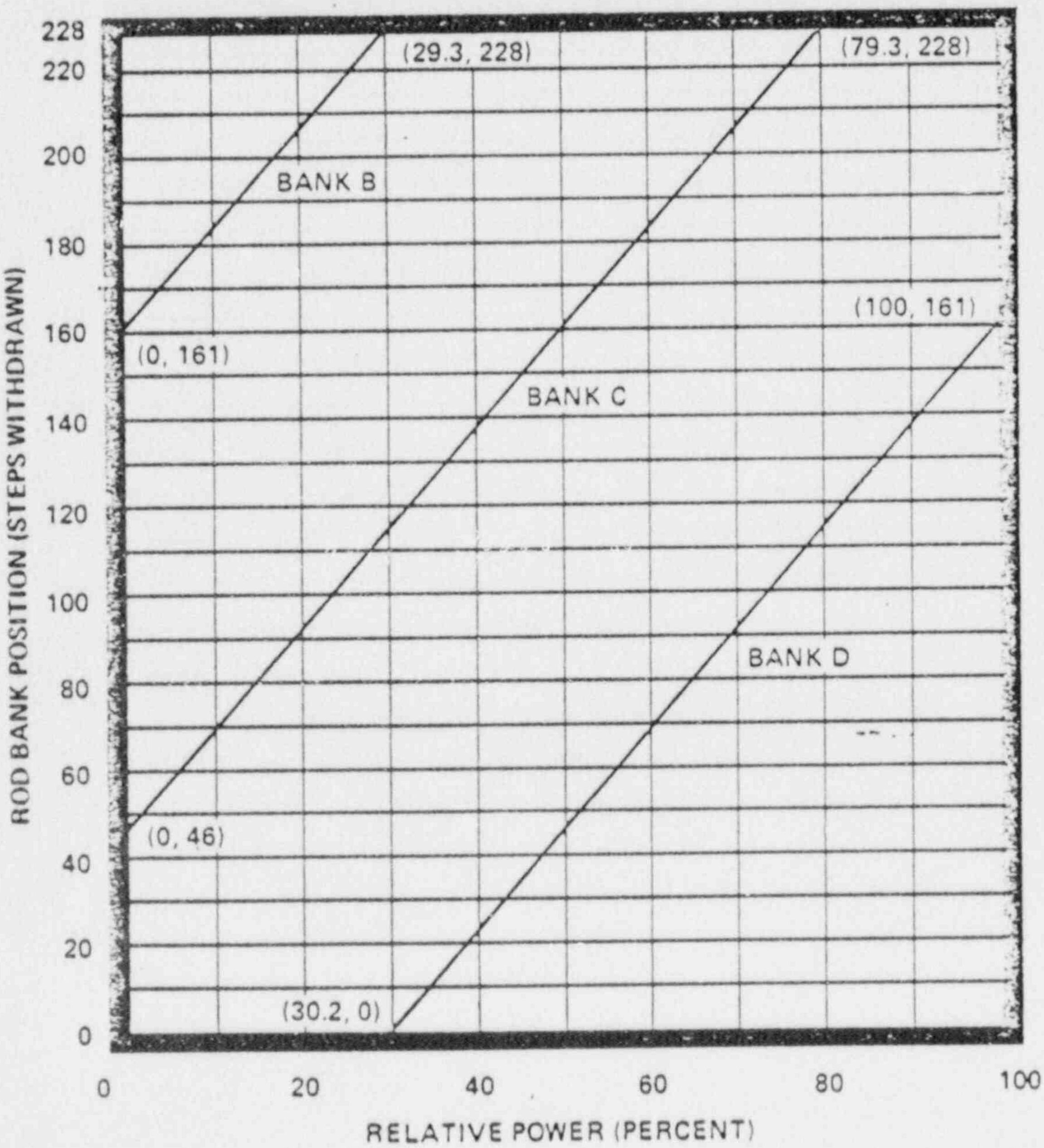
---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

FIGURE 3.1-3

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER



3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) about the target flux difference:

a. ± 5% for core average accumulated burnup of less than or equal to 3000 MWD/MTU; and

b. + 3%, -12% for core average accumulated burnup of greater than 3000 MWD/MTU.

The indicated AFD may deviate outside the above required target band 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 above required 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.

---

*See Special Test Exceptions Specification 3.10.2.

**Surveillance testing of the Power Range Neutron Flux Channel may be performed (below 90% of RATED THERMAL POWER) 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.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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 and the cumulative penalty deviation has been reduced to less than 1 hour in the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated 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 until the AFD Monitor Alarm is updated after restoration to OPERABLE status.

b. Monitoring and logging the indicated 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 AFD Monitor Alarm is inoperable. The logged values of the indicated 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 target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

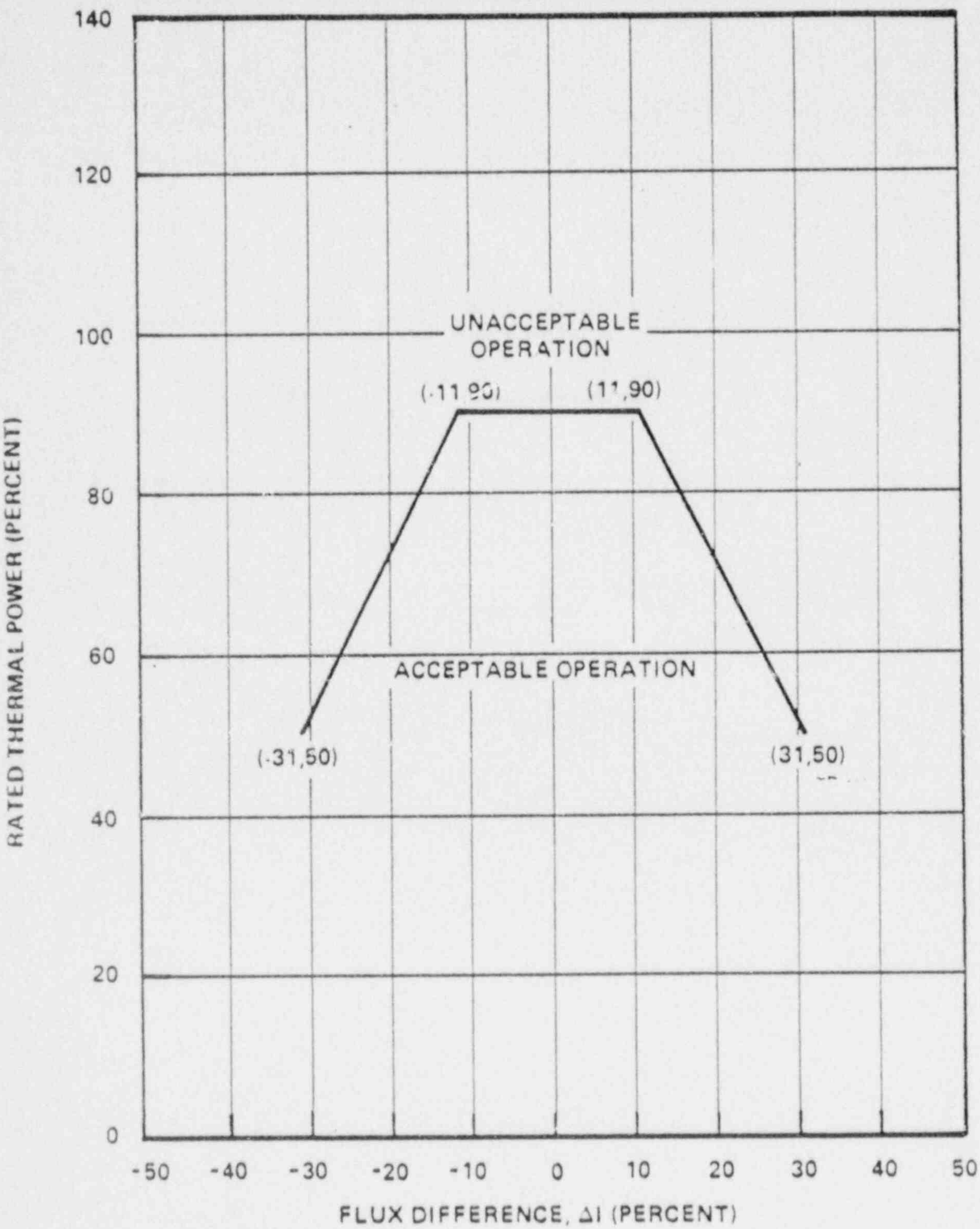
b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

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% 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:

$$F_Q(Z) \leq \frac{2.30}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq 4.60 [K(Z)] \text{ for } P \leq 0.5$$

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

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

APPLICABILITY: MODE 1.

ACTION:

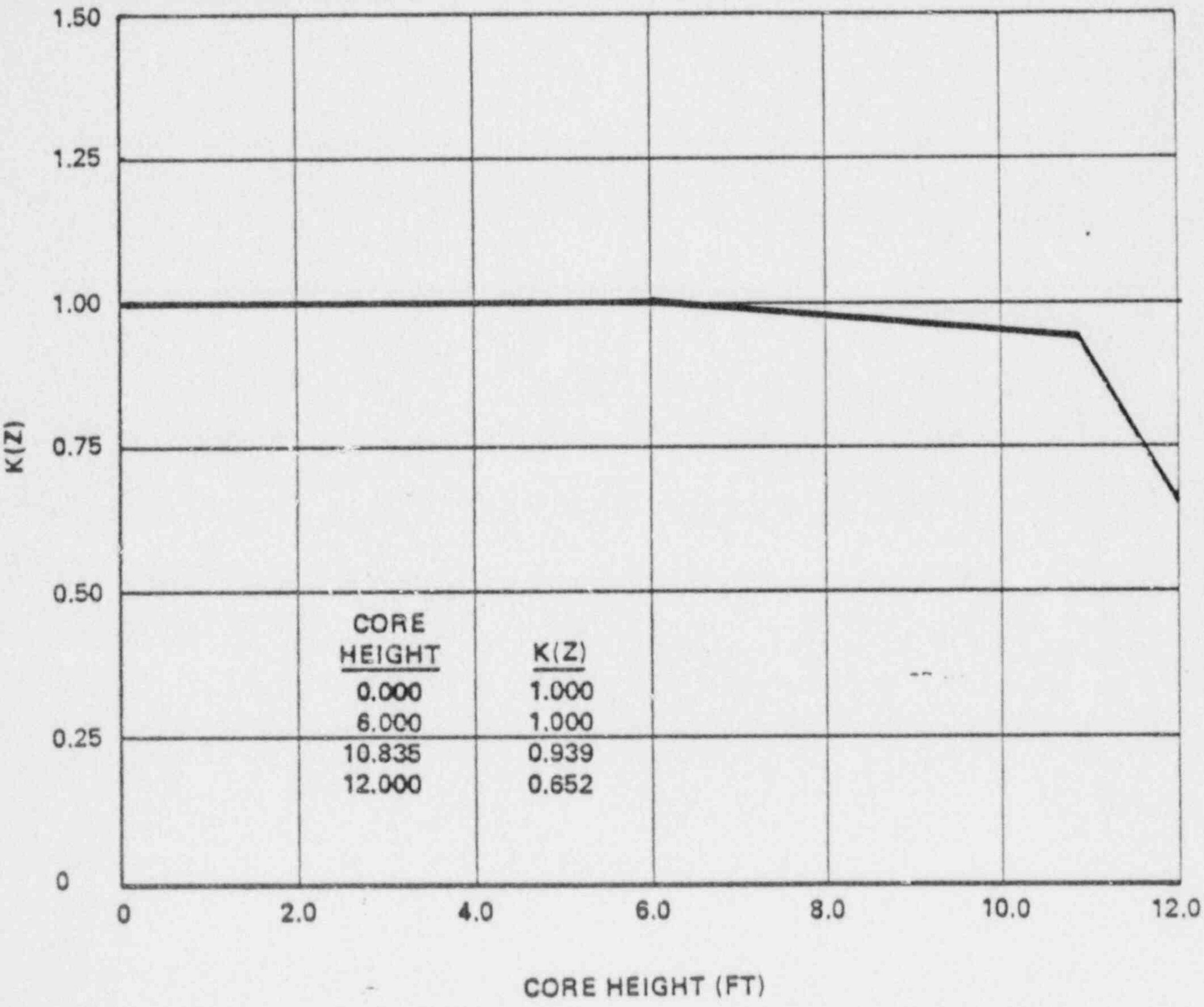
With $F_Q(Z)$ exceeding its limit:

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 ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit; and

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.

FIGURE 3.2-2

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



SURVEILLANCE REQUIREMENTS

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 before exceeding 75% of RATED THERMAL POWER following each fuel loading.

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 Effective Full Power Days (EFPD), whichever occurs first.

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} limits used in the Constant Axial Offset Control analysis 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 6.8.1.6;

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core planes 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 for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR-$F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55\,[1 + 0.3(1 - P)]$$

Where

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Within 4 hours either:

   1. Restore $F^N_{\Delta H}$ to within the above limit, 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.

b. Within 24 hours of initially being outside the above limit, verify through incore flux mapping that $F^N_{\Delta H}$ has been restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

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 $F^N_{\Delta H}$ is demonstrated, through incore flux mapping to be within its limit 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.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

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.

c. The measured $F^N_{\Delta H}$ shall be increased by 4% for measurement uncertainty.

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.

*See Special Test Exceptions Specification 3.10.2.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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:

      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, 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:

   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.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. 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

3. 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.

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% 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 two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits:

a. Reactor Coolant System T_{avg} (TI-0412, TI-0422, TI-0432, TI-0442), $\leq$ 591°F

b. Pressurizer Pressure (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), $\geq$ 2224 psig*

c. Reactor Coolant System Flow (FI-0414, FI-0415, FI-0416, FI-0424, FI-0425, FI-0426, FI-0434, FI-0435, FI-0436, FI-0444, FI-0445, FI-0446) $\geq$396,198 gpm **

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 Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once per 12 hours. RCS flow rate shall be monitored for degradation at least once per 12 hours. In the event of flow degradation, RCS flow rate shall be determined by precision heat balance within 7 days of detection of flow degradation.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at each fuel loading and at least once per 18 months.

4.2.5.3 After each fuel loading, the RCS flow rate shall be determined by precision heat balance prior to operation above 75% RATED THERMAL POWER. The RCS flow rate shall also be determined by precision heat balance at least once per 18 months. Within 7 days prior to performing the precision heat balance flow measurement, the instrumentation used for performing the precision heat balance shall be calibrated. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement.

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

**Includes a 3.5% flow measurement uncertainty.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION 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 their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 2.2-1 and with response times within their limit value.

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 the 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[a], 4[a], 5[a]	11
2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)					
a. High Setpoint	4	2	3	1, 2	2[b]
b. Low Setpoint	4	2	3	1[d], 2	2[b]
3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	4	2	3	1, 2	2[b]
4. Power Range, Neutron Flux, High Negative Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	4	2	3	1, 2	2[b]
5. Intermediate Range, Neutron Flux (NI-0035B,D&E NI-0036B,D&G)					
a. Power Operation	2	1	2	1[d]	3
b. Startup	2	1	2	2	3
6. Source Range, Neutron Flux (NI-0031B,D&E, NI-0032B,D&G)					
a. Startup	2	1	2	2[c]	4
b. Shutdown	2	1	2	3[j], 4, 5	5

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Overtemperature ΔT (TDI-0411C, TDI-0421C, TDI-0431C, TDI-0441C)	4	2	3	1, 2	6^b
8. Overpower ΔT (TDI-0411B, TDI-0421B, TDI-0431B, TDI-0441B)	4	2	3	1, 2	6^b
9. Pressurizer Pressure--Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	4	2	3	1^f	6^b
10. Pressurizer Pressure--High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	4	2	3	1, 2	6^b
11. Pressurizer Water Level--High* (LI-0459A, LI-0460A, LI-0461A)	3	2	2	1^f	6^b
12. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8) (LOOP 1: FI-0414, FI-0415, FI-0416; LOOP 2: FI-0424, FI-0425, FI-0426; LOOP 3: FI-0434, FI-0435, FI-0436; LOOP 4: FI-0444, FI-0445, FI-0446)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1^h	6^b

*See Specification 3.3.3.6

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1[f]	6[b]
(LOOP1 LOOP2 LOOP3 LOOP4 FI-0414 FI-0424 FI-0434 FI-0444 FI-0415 FI-0425 FI-0435 FI-0445 FI-0416 FI-0426 FI-0436 FI-0446)					
13. Steam Generator Water Level--Low-Low*	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6[b,g]
(LOOP1 LOOP2 LOOP3 LOOP4 LI-0517 LI-0527 LI-0537 LI-0547 LI-0518 LI-0528 LI-0538 LI-0548 LI-0519 LI-0529 LI-0539 LI-0549 LI-0551 LI-0552 LI-0553 LI-0554)					
14. Undervoltage--Reactor Coolant Pumps	4-2/bus	2[i]	3	1[f]	6[b]
15. Underfrequency--Reactor Coolant Pumps	4-2/bus	2[i]	3	1[f]	6[b]
16. Turbine Trip					
a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163)	3	2	2	1[e]	6[b]
b. Turbine Stop Valve Closure	4	4	1	1[e]	12[b]
17. Safety Injection Input from ESF	2	1	2	1, 2	10

*See Specification 3.3.3.6

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. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6 (NI-0035B,D&E, NI-0036B,D&G)	2	1	2	2^c	8
b. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	4	2	3	1	8
c. Power Range Neutron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	4	2	3	1	8
d. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	4	2	3	1,2	8
e. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	2	1	2	1	8
19. Reactor Trip Breakers	2	1	2	1, 2	10, 13
	2	1	2	3^a, 4^a, 5^a	11
20. Automatic Trip and Interlock	2	1	2	1, 2	10
Logic	2	1	2	3^a, 4^a, 5^a	11

TABLE NOTATIONS

a When the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

b The provisions of Specification 3.0.4 are not applicable.

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

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

e Above the P-9 (Reactor Trip on Turbine Trip Interlock) setpoint.

f Above the P-7 (Low Power Reactor Trip Block) setpoint.

g The applicable Modes and Action Statement for these channels noted in Table 3.3-3 are more restrictive and therefore, applicable.

h Above the P-8 (Single Loop Loss of Flow) Setpoint.

i Trip logic consists of under voltage/under frequency for Reactor Coolant Pumps 1 or 2 and 3 or 4.

j The Source Range High Flux at Shutdown Alarm may be blocked during reactor start up in accordance with approved procedures.

ACTION STATEMENTS

ACTION 1 - 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 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/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER 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 (Continued)

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, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) 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 OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - a. 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 System breakers, suspend all operations involving positive reactivity changes and verify valves 1208-U4-175, 1208-U4-177, 1208-U4-183, and 1208-U4-176 are closed and secured in position within the next hour.

b. With no channels OPERABLE, open the Reactor Trip Breakers, suspend all operations involving positive reactivity changes, and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and every 12 hours thereafter. Verify valves 1208-U4-175, 1208-U4-177, 1208-U4-183, and 1208-U4-176 are closed and secured in position within 4 hours and verified to be closed and secured in position every 14 days.

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 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - (Not used)

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

ACTION 9 - (Not used).

ACTION 10 - 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 4.3.1.1, provided the other channel is OPERABLE.

ACTION 11 - 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 System breakers within the next hour.

ACTION 12 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 13 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 10. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

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(14)	N.A.	1, 2, 3[a], 4[a], 5[a]
2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q(17)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1[d], 2
3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	R(4)	Q(17)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	R(4)	Q(17)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux (NI-0035B,D&E, NI-0036B,D&G)	S	R(4, 5)	S/U(1)	N.A.	N.A.	1[d], 2
6. Source Range, Neutron Flux (NI-0031B,D&E, NI-0032B,D&G)	S	R(4, 5)	S/U(1),Q(9,17)	N.A.	N.A.	2[c], 3, 4, 5
7. Overtemperature ΔT (TDI-0411C, TDI-0421C, TDI-0431C, TDI-0441C)	S	R(12)	Q(17)	N.A.	N.A.	1, 2

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
8. Overpower ΔT (TDI-0411B, TDI-0421B, TDI-0431B, TDI-0441B)	S	R	Q(17)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	S	R	Q(17)	N.A.	N.A.	1^e
10. Pressurizer Pressure--High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	S	R	Q(17)	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High* (LI-0459A, LI-0460A, LI-0461A)	S	R	Q(17)	N.A.	N.A.	1^e
12. Reactor Coolant Flow--Low	S	R	Q(17)	N.A	N.A.	1^e

(LOOP1	LOOP2	LOOP3	LOOP4
FI-0414	FI-0424	FI-0434	FI-0444
FI-0415	FI-0425	FI-0435	FI-0445
FI-0416	FI-0426	FI-0436	FI-0446)

*See Specification 4.3.3.6

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(17, 18)	N.A.	N.A.	1, 2
(LOOP1 LOOP2 LOOP3 LOOP4 LI-0517 LI-0527 LI-0537 LI-0547 LI-0518 LI-0528 LI-0538 LI-0548 LI-0519 LI-0529 LI-0539 LI-0549 LI-0551 LI-0552 LI-0553 LI-0554)						
14. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q(17)	N.A.	1^e
15. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q(17)	N.A.	1^e
16. Turbine Trip						
a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163)	N.A.	R	S/U,(1, 10)	N.A.	N.A.	1^b
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1^b
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6 (NI-0035B,D&E, NI-0036B,D&G)	N.A.	R(4)	R	N.A.	N.A.	2^c

* See Specification 4.3.3.6

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
18. Reactor Trip System Interlocks (Continued)						
b. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C)	N.A.	R(4)	R	N.A.	N.A.	1
d. Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	N.A.	R	R	N.A.	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3[a], 4[a], 5[a]
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3[a], 4[a], 5[a]
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(15),R(16)	N.A.	1, 2, 3[a], 4[a], 5[a]

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

a When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

b Above P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.

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

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

e Above P-7 (Low Power Reactor Trip Block) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. 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. For the purpose of this surveillance requirement, monthly shall mean at least once per 31 EFPD.

(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. This is the determination of the response of the excore power range detectors to the incore measured axial power distribution to generate setpoints for the CHANNEL CALIBRATION. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purpose of this surveillance requirement, quarterly shall mean at least once per 92 EFPD.

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

(8) Not used

(9) Quarterly surveillance in MODES 3[a], 4[a], and 5[a] 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 window. Quarterly surveillance shall include verification of the Source Range High Flux at Shutdown Alarm Setpoint of less than or equal to 3.16 times background.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the OPERABILITY of the Undervoltage and Shunt trip of the Reactor Trip Breaker.

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

(13) Not used

(14) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(15) Local manual shunt trip prior to placing breaker in service.

(16) Automatic undervoltage trip.

(17) Each channel shall be tested at least every 92 days on a STAGGERED TEST BASIS.

(18) The surveillance frequency and/or MODES specified for these channels in Table 4.3-2 are more restrictive and, therefore, applicable.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-2 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-3 and with response times within their limit value.

APPLICABILITY: As shown in Table 3.3-2.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-3, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-3, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-3, and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-2 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $Z + R \pm S \leq TA$

   Where:

   Z = The value from Column Z of Table 3.3-3 for the affected channel,

   R = The "as measured" value (in percent span) of rack error for the affected channel,

   S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 3.3-3 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 3.3-3 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-2.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS 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 3.3-2.

TABLE 3.3-2

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, Component Cooling Water, Control Room Emergency Mode Actuation, Start Diesel Generators, Containment Cooling Fans, Nuclear Servi[illegible] Cooling Water, Containment Isolatio[illegible], Containment Ventilation Isolation, and Auxiliary Feedwater Motor-Driven Pumps).					
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-1* (PI-0934, PI-0935, PI-0936)	3	2	2	1, 2, 3, 4	15^d
d. Pressurizer Pressure--Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	4	2	3	1, 2, 3^a	20^d
e. Steam Line Pressure--Low*	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3^a	15^d

(LOOP1	LOOP2	LOOP3	LOOP4
PI-0514A,B&C	PI-0524A&B	PI-0534A&B	PI-0544A,B&C
PI-0515A	PI-0525A	PI-0535A	PI-0545A
PI-0516A	PI-0526A	PI-0536A	PI-0546A)

*See Specification 3.3.3.6

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Manual Initiation	2	1 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-3* (PI-0934, PI-0935, PI-0936, PI-0937)	4	2	3	1, 2, 3, 4	17
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	19
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Functional Unit 1. above for all Safety Injection initiating functions and requirements.				

*See Specification 3.3.3.6

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Containment Ventilation Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4, 6^c	16, 18
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4, 6^c	16, 18
3) Safety Injection	See Functional Unit 1. above for all Safety Injection initiating functions and requirements.				
4) Containment Radiation; Area Low Range (RE-0002, RE-0003) and Containment Ventilation Radiation (RE-2565A - Particulate, RE-2565B - Iodine, RE-2565C - Gaseous)	3	1	2^g	1, 2, 3, 4, 6^c	16, 18
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	2/steam line	1/steam line	1/operating steam line	1, 2^f, 3^f	24
2) System	2	1	2	1, 2^f, 3^f	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2^f, 3^f	22

TABLE 3.3-2 (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)					
c. Containment Pressure--High-2 * (PI-0934, PI-0935, PI-0936)	3	2	2	1, 2[f], 3[f]	15[d]
d. Steam Line Pressure--Low*	3/steam line	2/steam line any steam line	2/steam line	1, 2[f], 3[a,f]	15[d]
(LOOP1 PI-0514A,B&C PI-0515A PI-0516A; LOOP2 PI-0524A&B PI-0525A PI-0526A; LOOP3 PI-0534A&B PI-0535A PI-0536A; LOOP4 PI-0544A,B&C, PI-0545A, PI-0546A)					
e. Steam Line Pressure - Negative Rate--High *	3/steam line	2/steam line any steam line	2/steam line	3[b,f]	15[d]
(LOOP1 PI-0514A,B&C PI-0515A PI-0516A; LOOP2 PI-0524A&B PI-0525A PI-0526A; LOOP3 PI-0534A&B PI-0535A PI-0536A; LOOP4 PI-0544A,B&C PI-0545A PI-0546A)					
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	25
b. Low RCS Tavg Coincident with Reactor Trip**					
1. Low RCS T_{avg}	4	2	3	1, 2	20[d]
2. Reactor Trip, P-4	See Functional Unit 9b for P-4 requirements.				

*See Specification 3.3.3.6

**Feedwater isolation only. Turbine trip occurs on reactor trip.

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. Turbine Trip and Feedwater Isolation (Continued)					
c. Steam Generator Water Level--High-High (P-14) *	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm.gen. in each operating stm. gen.	1, 2	20[d]
(LOOP1 LOOP2 LOOP3 LOOP4 LI-0517 LI-0527 LI-0537 LI-0547 LI-0518 LI-0528 LI-0538 LI-0548 LI-0519 LI-0529 LI-0539 LI-0549 LI-0551 LI-0552 LI-0553 LI-0554)					
d. Safety Injection	See Functional Unit 1 above for all Safety Injection initiating functions and requirements.				
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22

* See Specification 3.3.3.6

TABLE 3.3-2 (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 (Continued)					
b. Stm. Gen. Water Level-- Low-Low*					
(LOOP1 LOOP2 LOOP3 LOOP4					
LI-0517 LI-0527 LI-0537 LI-0547					
LI-0518 LI-0528 LI-0538 LI-0548					
LI-0519 LI-0529 LI-0539 LI-0549					
LI-0551 LI-0552 LI-0553 LI-0554)					
1) Start Motor-Driven Pumps	4/stm. gen.	2/stm. gen. in any operating stm.gen	3/stm. gen. in each operating stm. gen.	1, 2, 3	20[d]
2) Start Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	20[d]
c. Safety Injection Start Motor-Driven Pumps	See Functional Unit 1. above for all Safety Injection initiating functions and requirements.				
d. Loss of or Degraded 4.16 kV ESF Bus Voltage					
i Start Motor-Driven Pumps	4/train	2/train	3/train	1, 2, 3	23
ii Start Turbine-Driven Pump	4/train	2 from either train	3/train	1, 2, 3	23

*See Specification 3.3.3.6

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTI
e. Trip of All Main Feedwater Pumps, Start Motor-Driven Pumps	2	2	2	1, 2	23
f. Manual Initiation	3^h	1/pump	1/pump	1, 2, 3	23
7. Semi-Automatic Switchover to Containment Emergency Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. RWST Level--Low-Low Coincident with Safety Injection* (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A)	4	2	3	1, 2, 3, 4	17
	See Functional Unit 1. above for all Safety Injection initiating functions and requirements.				
8. Loss of Power to 4.16 kV ESF Bus					
a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20^d
b. 4.16 kV ESF Bus Undervoltage-Degraded Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20^d
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11 (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A)	3	2	2	1, 2, 3	21
b. Reactor Trip, P-4	2	1	2	1, 2, 3	23

*See Specification 3.3.3.6

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Control Room Ventilation Emergency Mode Actuation					
a. Manual Initiation	2	1	1	1, 2, 3, 4, 5^e, 6^e	26^d
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4, 5^e, 6^e	26^d
c. Safety Injection	See Functional Unit 1 above for all Safety Injection initiating functions and requirements.				
d. Intake Radiogas Monitor (RE-12116, RE-12117)	2	1	2	1, 2, 3, 4, 5^e,6^e	26^d
11. Fuel Handling Building Post Accident Ventilation Actuation (Common System)					
a. Manual Initiation	2	1	1	i	j
b. Fuel Handling Building Exhaust Duct Radiation Signal (ARE-2532 A&B ARE-2533 A&B)	4	1	1	i	j
c. Automatic Actuation Logic and Actuation Relays	4	1	1	i	j

TABLE 3.3-2 (Continued)

TABLE NOTATIONS

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

b Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

c During movement of irradiated fuel or movement of loads over irradiated fuel within containment.

d The provisions of Specification 3.0.4 are not applicable.

e During movement of irradiated fuel or movement of loads over irradiated fuel.

f Not applicable if one main steam isolation valve and associated bypass isolation valve per steamline is closed.

g Containment Ventilation Radiation (RE-2565) is treated as one channel and is considered OPERABLE if the particulate (RE-2565A) and iodine monitors (RE-2565B) are OPERABLE or the noble gas monitor (RE-2565C) is OPERABLE.

h Manual initiation of Auxiliary Feedwater is accomplished via the pump handswitches.

i Whenever irradiated fuel is in ~~the~~ either storage pool.

j For actions associated with inoperable instrumentation, follow actions specified in Specification 3.9.12.

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 less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9 (Mode 6).

TABLE 3.3-2 (Continued)

ACTION STATEMENTS (Continued)

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 one less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are closed within 24 hours.

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 is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 21 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, within 1 hour determine by observation of the associated permissive status light(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 Minimum Channels OPERABLE requirement, 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.

TABLE 3.3-2 (Continued)

ACTION STATEMENTS (Continued)

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

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.

ACTION 26 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room Emergency Ventilation System in the Emergency mode. With two channels inoperable, within 1 hour initiate and maintain operation of the Control Room Emergency Ventilation System in the Emergency mode.

ACTION 27 - Not Used.

ACTION 28 - Not Used.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Component Cooling Water, Control Room Emergency Mode Actuation, Start Diesel Generators, Containment Cooling Fans, Nuclear Service Cooling Water, Containment Isolation, Containment Ventilation Isolation, and Auxiliary Feedwater Motor-Driven Pumps)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High 1 (PI-0934, PI-0935, PI-0936)	2.7	0.71	1.67	$\leq$ 3.8 psig	$\leq$ 4.4 psig
d. Pressurizer Pressure--Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	13.1	10.71	1.67	$\geq$ 1870 psig***	$\geq$ 1860 psig
e. Steam Line Pressure--Low	13.0	10.71	1.67	$\geq$ 585 psig*	$\geq$ 570 psig

(LOOP1	LOOP2	LOOP3	LOOP4
PI-0514A,B&C	PI-0524A&B	PI-0534A&B	PI-0544A,B&C
PI-0515A	PI-0525A	PI-0535A	PI-0545A
PI-0516A	PI-0526A	PI-0536A	PI-0546A)

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3 (PI-0934, PI-0935, PI-0936, PI-0937)	3.1	0.71	1.67	$\leq$ 21.5 psig	$\leq$ 22.4 psig

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Functional Unit 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Containment Ventilation Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Functional Unit 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
4) Containment Radiation					
i. Area Low Range (RE-0002, RE-0003)	N.A.	N.A.	N.A.	15 mr/hr(a) 100 mr/hr(b)	N.A.
ii. Ventilation Particulate Activity (RE-2565A)	N.A.	N.A.	N.A.	C	N.A.
iii. Ventilation Iodine Activity (RE-2565B)	N.A.	N.A.	N.A.	C	N.A.
iv. Ventilation Gaseous Activity (RE-2565C)	N.A.	N.A.	N.A.	C	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-2 (PI-0934, PI-0935, PI-0936)	3.1	0.71	1.67	≤14.5 psig	≤15.4 psig
d. Steam Line Pressure--Low	13.0	10.71	1.67	≥585 psig*	≥570 psig
(LOOP1: PI-0514A,B&C, PI-0515A, PI-0516A; LOOP2: PI-0524A&B, PI-0525A, PI-0526A; LOOP3: PI-0534A&B, PI-0535A, PI-0536A; LOOP4: PI-0544A,B&C, PI-0545A, PI-0546A)					
e. Steam Line Pressure - Negative Rate--High	3.0	0.50	0	≤100 psi**	≤125 psi
(LOOP1: PI-0514A,B&C, PI-0515A, PI-0516A; LOOP2: PI-0524A&B, PI-0525A, PI-0526A; LOOP3: PI-0534A&B, PI-0535A, PI-0536A; LOOP4: PI-0544A,B&C, PI-0545A, PI-0546A)					
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Low RCS Tavg Coincident with Reactor Trip##					
1. Low RCS T_{avg}	4.0	0.82	0.87	≥564°F	≥561.5°F
2. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (Continued)					
c. Steam Generator Water Level--High-High (P-14)	5.1	2.18	1.67	≤78.0% of narrow range	≤79.9% of narrow range instrument instrument span.

(LOOP1	LOOP2	LOOP3	LOOP4
LI-0517	LI-0527	LI-0537	LI-0547
LI-0518	LI-0528	LI-0538	LI-0548
LI-0519	LI-0529	LI-0539	LI-0549
LI-0551	LI-0552	LI-0553	LI-0554)

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. Safety Injection	See Functional Unit 1 above for all Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level--Low-Low					

(LOOP1	LOOP2	LOOP3	LOOP4
LI-0517	LI-0527	LI-0537	LI-0547
LI-0518	LI-0528	LI-0538	LI-0548
LI-0519	LI-0529	LI-0539	LI-0549
LI-0551	LI-0552	LI-0553	LI-0554)

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Start Motor-Driven Pumps	18.5	17.18	1.67	≥18.5% of# narrow range instrument span.	≥17.8% of narrow range instrument span.
2. Start Turbine-Driven Pump	18.5	17.18	1.67	≥18.5% of# narrow range instrument span.	≥17.8% of narrow range instrument span.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
c. Safety Injection Start Motor-Driven Pumps	See Functional Unit 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
d. Loss of or degraded 4.16 kV ESF Bus Voltage					
1. Start Motor-Driven Pumps	N.A.	N.A.	N.A.	Degraded ≥3746 volts with ≤20 s time delay Loss ≥2975 volts with a <0.8s time delay	Degraded ≥3683 volts with ≤20 s time delay Loss ≥2912 volts with a <0.8s time delay
2. Turbine Driven Pump	N.A.	N.A.	N.A.	Degraded ≥3746 volts with ≤20 s time delay Loss ≥2975 volts with a <0.8s time delay	Degraded ≥3683 volts with ≤20 s time delay Loss ≥2912 volts with a <0.8s time delay
e. Trip of All Main Feedwater Pumps, Start Motor-Driven Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
f. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
7. Semi-Automatic Switchover to Containment Emergency Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
7. Semi-Automatic Switchover to Containment Emergency Sump (Continued)					
b. RWST Level--Low-Low Coincident With Safety Injection (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A)	3.5	0.71	1.67	≥275.3 in. from tank base (≥39.1% of instrument span)	≥264.9 in. from tank base (≥37.4% of instrument span)
8. Loss of Power to 4.16 kV ESF Bus					
a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage	N.A.	N.A.	N.A.	≥2975 volts with a ≤ 0.8 second time delay.	≥2912 volts with a ≤ 0.8 second time delay.
b. 4.16 kV ESF Bus Undervoltage-Degraded Voltage	N.A.	N.A.	N.A.	≥3746 volts with a <20 second time delay.	≥3683 volts with a <20 second time delay.
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11 (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	N.A.	N.A.	N.A.	≤ 1970 psig	≤ 1980 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
10. Control Room Ventilation Emergency Mode Actuation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Safety Injection	See Functional Unit 1 above for all Safety Injection Trip Setpoints and Allowable Values.				
d. Intake Radiogas monitor (RE-12116, RE-12117)	N.A.	N.A.	N.A.	$\leq$ 3X background	N.A.
11. Fuel Handling Building ~~Post-Accident~~ Ventilation ~~System~~ Actuation (Common System					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Fuel Handling Building Exhaust Duct Radiation Signal (ARE-2532A&B, ARE-2533A&B)	N.A.	N.A.	N.A.	C	N.A.
c. Automatic Actuation Logic And Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

***Until resolution of the Veritrak transmitter uncertainty issue this setpoint will be set at $\geq$1885 psig.

#Until resolution of the Veritrak transmitter uncertainty issue the setpoint will be set at $\geq$22.5% (Unit 1) and $\geq$ 23.5% (Unit 2) of narrow range instrument span.

##Feedwater isolation only. Turbine trip occurs on reactor trip.

[a]During refueling operations.

[b]During power operation. This is an initial setpoint only. The trip setpoint will be set at 50 times background level. Background level should be determined at or near the end of the first fuel cycle.

[c]Setpoints will not exceed the limits of Specification 3.11.2.1.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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, Component Cooling Water, Control Room Emergency Mode Actuation, Start Diesel Generators, Containment Cooling Fans, Nuclear Service Cooling Water, Containment Isolation, Containment Ventilation Isolation, and Auxiliary Feedwater Motor-driven Pumps).								
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-1 (PI-0934, PI-0935, PI-0936)*	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

*See Specification 4.3.3.6

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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 SURVEILLANC IS REQUIRED
1. (Continued)								
e. Steam Line Pressure-Low*	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
(LOOP1 PI-0514A,B&C PI-0515A PI-0516A; LOOP2 PI-0524A&B PI-0525A PI-0526A; LOOP3 PI-0534A&B PI-0535A PI-0536A; LOOP4 PI-0544A,B&C PI-0545A PI-0546A)								
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-3* (PI-0934, PI-0935, PI-0936, PI-0937)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	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(i)	Q	1, 2, 3, 4

*See Specification 4.3.3.6.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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
3. Containment Isolation (Continued)								
3) Safety Injection	See Functional Unit 1. above for all Safety Injection Surveillance Requirements.							
b. Containment Ventilation Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3 4, 6#
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4, 6#
3) Safety Injection	See Functional Unit 1. above for all Safety Injection Surveillance Requirements.							
4) Containment Radiation								
i. Area Low Range (RE-0002, RE-0003)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4, 6#
ii. Ventilation Particulate Activity (RE-2565A)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4, 6#
iii. Ventilation Iodine Activity (RE-2565B)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4, 6#
iv. Ventilation Gaseous Activity (RE-2565C)	S	R	M	N.A.	N.A.	N.A	N.A.	1, 2, 3 4, 6#

#During movement of irradiated fuel or movement of loads over irradiated fuel within containment.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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 Initiation	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-2* (PI-0934, PI-0935, PI-0936)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low*	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
(LOOP1	LOOP2	LOOP3	LOOP4					
PI-0514A,B&C PI-0515A PI-0516A	PI-0524A&B PI-0525A, PI-0526A	PI-0534A&B PI-0535A PI-0536A	PI-0544A,B&C PI-0545A PI-0546A)					
e. Steam Line Pressure-Negative Rate-High*	S	R	M	N.A.	N.A.	N.A.	N.A.	3
(LOOP1	LOOP2	LOOP3	LOOP4					
PI-0514A,B&C PI-0515A PI-0516A	PI-0524A&B PI-0525A PI-0526A	PI-0534A&B PI-0535A PI-0536A	PI-0544A,B&C PI-0545A PI-0546A)					

*See Specification 4.3.3.6

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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
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. Low RCS Tavg Coincident with Reactor Trip*								
1. Low RCS T_{avg}	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
2. Reactor Trip, P-4		See functional unit 9b for P-4 Surveillance requirements.						
c. Steam Generator Water Level-High-High (P-14) **	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2

(LOOP1	LOOP2	LOOP3	LOOP4
LI-0517	LI-0527	LI-0537	LI-0547
LI-0518	LI-0528	LI-0538	LI-0548
LI-0519	LI-0529	LI-0539	LI-0549
LI-0551	LI-0552	LI-0553	LI-0554)

d. Safety Injection See Functional Unit 1 above for all Safety Injection Surveillance Requirements.

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*Feedwater isolation only. Turbine trip occurs on reactor trip.

** See Specification 4.3.3.6

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
6. Auxiliary Feedwater								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
b. Steam Generator Water Level-Low-Low* (LOOP1 LOOP2 LOOP3 LOOP4 / LI-0517 LI-0527 LI-0537 LI-0547 / LI-0518 LI-0528 LI-0538 LI-0548 / LI-0519 LI-0529 LI-0539 LI-0549 / LI-0551 LI-0552 LI-0553 LI-0554)								
1) Start Motor-Driven Pumps	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
2) Start Turbine Driven Pump	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
c. Safety Injection Start Motor-Driven Pumps	See Functional Unit 1. above for all Safety Injection Surveillance Requirements.							
d. Loss of or Degraded 4.16 kV ESF Bus Voltage	N.A.	R	M	N.A.	N.A.	N.A.	N.A	1, 2, 3
e. Trip of All Main Feed-Water Pumps, Start Motor-Driven Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2
f. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3

*See Specification 4.3.3.6

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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
7. Semi-Automatic Switchover to Containment Emergency Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. RWST Level-Low-Low* Coincident With Safety Injection (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
8. Loss of Power to 4.16 kV ESF Bus								
a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4.16 kV ESF Bus Undervoltage-Degraded Voltage	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

*See Specification 4.3.3.6

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11 (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A)	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
10. Control Room Ventilation Emergency Mode Actuation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4, 5#, 6#
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3, 4, 5#, 6#

~~*See Specification 4.3.3.6~~
#During movement of irradiated fuel or movement of loads over irradiated fuel.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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
10. Control Room Ventilation Emergency Mode Actuation (Continued)								
c. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements							
d. Intake Radiogas Monitor (RE-12116, RE-12117)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4, 5#, 6#
11. Fuel Handling Building Post Accident Ventilation Actuation (Common System)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	(2)
b. Fuel Handling Building Exhaust Duct Radiation Signal (ARE-2532 A&B ARE-2533 A&B)	S	R	M	N.A.	N.A.	N.A.	N.A.	(2)
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	(2)

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Whenever irradiated fuel is in the storage pool.
During movement of irradiated fuel or movement of loads over irridated fuel.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-4 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-4.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-4, 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 3.3-4.

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 ANALOG CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Area (High Range) (RE-0005, RE-0006)	1	See Table 3.3-8	1, 2, 3*, 4*	100 R/hr	See Table 3.3-8
b. RCS Leakage Detection					
1) Gaseous Activity (RE-2562C)	1	1	1, 2, 3, 4	$\leq$ 2 x back-ground	29
2) Particulate Activity (RE-2562A)	1	1	1, 2, 3, 4	$\leq$ 2 x back-ground	29
2. Containment Ventilation	1	2^c	1, 2, 3, 4, 6^a	See Table 3.3-3	See Table 3.3-2
Area Low Range (RE-0002, RE-0003) Gaseous Activity (RE-2565C) Particulate Activity (RE-2565A) Iodine Activity (RE-2565B)					
3. Control Room Air Intake					
(RE-12116, RE-12117)	1	2	1, 2, 3, 4 5^b, 6^b	See Table 3.3-3	See Table 3.3-2

* The provisions of this specification are not applicable to Unit 2 until its initial entry into MODE 2.

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

a During movement of irradiated fuel or movement of loads over irradiated fuel within containment.

b During movement of irradiated fuel or movement of loads over irradiated fuel.

c RE-2565 is considered OPERABLE if the Particulate (RE-2565A) and Iodine (RE-2565B) Monitors are OPERABLE, or the noble gas monitor (RE-2565C) is OPERABLE.

ACTION STATEMENTS

ACTION 29 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, satisfy the ACTION requirements of Specification 3.4.6.1.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment Area (High Range) (RE-0005, RE-0006)	See Specification 4.3.3.6		M	1, 2, 3*, 4*
b. RCS Leakage Detection				
1) Gaseous Activity (RE-2562C)	S	R	M	1, 2, 3, 4
2) Particulate Activity (RE-2562A)	S	R	M	1, 2, 3, 4
2. Containment Ventilation Area Low Range (RE-0002, RE-0003) Gaseous Activity (RE-2565C) Particulate Activity (RE-2565A) Iodine Activity (RE-2565B)	See Table 4.3-2			
3. Control Room Air Intake (RE-12116, RE-12117)	See Table 4.3-2			

* The provisions of this specification are not applicable to Unit 2 until its initial entry into MODE 2.

MOVABLE INCORE DETECTORS

LIMITING CONDITION 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 two 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 within 24 hours prior to use by irradiating each detector used and determining the acceptability of its voltage curve 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}.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-5* shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.8.2 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 required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above required seismic monitoring instruments which is accessible during power operations and which is actuated during a seismic event greater than or equal to 0.01 g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 15 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 6.8.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

Each of the above seismic monitoring instruments which is actuated during a seismic event greater than or equal to 0.01g # but is not accessible during power operation shall be restored to OPERABLE status and a CHANNEL CALIBRATION performed the next time ~~the plant~~ unit 1 enters MODE 5 or below. A supplemental report shall then be prepared and submitted to the Commission with 14 days pursuant to Specification 6.8.2 describing the additional data from these instruments.

~~*The instrumentation may be shared with additional units at a common site provided seismic instrumentation and corresponding Technical Specifications meet the recommendations of Regulatory Guide 1.12, Revision 1, April 1974.~~

TABLE 3.3-5

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE	INSTRUMENT TAG NUMBER
1. Triaxial Time-History Accelerographs			
a. Free Field (500 ft from containment)	-1 to 1 g	1	AXT-19900
b. Unit 1 Containment Gallery (basemat)	-1 to 1 g	1	AXT-19901
c. Unit 1 Containment Operating Floor	-1 to 1 g	1	AXT-19902
d. Auxiliary Building Basemat	-1 to 1 g	1	AXT-19906
e. Unit 1 Containment Pressurizer Support	-1 to 1 g	1	AXT-19903
f. Auxiliary Building Level 1	-1 to 1 g	1	AXT-19905
2. Triaxial Peak Accelerographs			
a. Unit 1 Reactor Coolant Pump Motor (210 ft)	±10gHoriz/±5gVert	1	AXR-19910
b. Unit 1 Steam Generator (185 ft)	±2gHoriz/±5gVert	1	AXR-19911
c. Unit 1 NSCW Piping Outside Aux Bldg (220 ft)	-10g to +10g	1	AXR-19913
3. Triaxial Seismic Switch			
a. Unit 1 Containment Tendon Gallery (basemat)	**	1*	AXSH-19920
4. Triaxial Reponse-Spectrum Analyzer			
a. Control Room	Input: -1 to 1g Output: 0.03g to 9.99g	1*	AXA-19930
5. Triaxial Seismic Triggers			
a. Unit 1 Containment Tendon Gallery (basemat)	***	1*	AXSH-19922
b. Unit 1 Containment Operating Floor	***	1*	AXSH-19923

*With reactor control room indication.

**Triaxial seismic switch is set at the OBE acceleration level of 0.17g horizontal and 0.23g vertical.

***Triaxial seismic trigger is set at 0.01g all axes.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	ACCESSIBL DURING MODES
1. Triaxial Time-History Accelerographs				
a. Free Field (500 ft from containment)	M	R	SA	All
b. Unit 1 Containment Gallery (basemat)	M	R	SA	All
c. Unit 1 Containment Operating Floor	M	R	SA	All
d. Auxiliary Building Basemat	M	R	SA	All
e. Unit 1 Containment Pressurizer Support	M	R	SA	5, 6
f. Auxiliary Building Level 1	M	R	SA	All
2. Triaxial Peak Accelerographs				
a. Unit 1 Reactor Coolant Pump Motor (210 ft)	N.A.	R	N.A.	5, 6
b. Unit 1 Steam Generator (185 ft)	N.A.	R	N.A.	5,
c. Unit 1 NSCW Piping Outside Aux. Bldg. (220 ft)	N.A.	R	N.A.	All
3. Triaxial Seismic Switches				
a. Unit 1 Containment Tendon Gallery (basemat)	M	R	SA	All
4. Triaxial Response-Spectrum Analyzer				
a. Control Room*	M	R	N.A.	All
5. Triaxial Seismic Triggers				
a. Unit 1 Containment Tendon Gallery (basemat)	M	R	SA	All
b. Unit 1 Containment Operating Floor	M	R	SA	All

*With reactor control room indications.

METEOROLOGICAL INSTRUMENTATION (Common System)

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-6 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 6.8.2 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 meteorological monitoring instrumentation channels shown in Table 3.3-6 shall be demonstrated OPERABLE by the performance of:

a. A Daily CHANNEL CHECK, and

b. A Semiannual CHANNEL CALIBRATION.

TABLE 3.3-6

METEOROLOGICAL MONITORING INSTRUMENTATION *

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. Wind Speed		
a. Lower, Primary Tower	Nominal Elev. 10 m	1
b. Upper, Primary Tower	Nominal Elev. 60 m	1
2. Wind Direction		
a. Lower, Primary Tower	Nominal Elev. 10 m	1
b. Upper, Primary Tower	Nominal Elev. 60 m	1
3. Air Temperature - ΔT		
a. ΔT, Primary Tower	Nominal Elev. 10m-60m	1

* This instrumentation is common to Units 1 and 2.

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5.1 The Remote Shutdown System monitoring instrumentation channels shown in Table 3.3-7 shall be OPERABLE.

3.3.3.5.2 The Remote Shutdown System transfer switches and controls of system components required for safety grade 1) reactivity control, 2) RCS pressure control, 3) decay heat removal via auxiliary feedwater flow, steam generator atmospheric dump valve flow, and the RHR system, 4) RCS inventory control via charging flow, and 5) safety support systems required for the above functions shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

b. With one or more Remote Shutdown System transfer switches, or control circuits inoperable, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel in Table 3.3-7 shall be demonstrated OPERABLE:

a. Every 31 days by performance of a CHANNEL CHECK, and

b. Every 18 months by performance of a CHANNEL CALIBRATION.

4.3.3.5.2 Each Remote Shutdown System transfer switch and control circuit as required by Specification 3.3.3.5.2 shall be demonstrated OPERABLE at least once per 18 months by verifying its capability to perform its intended function(s).

TABLE 3.3-7

REMOTE SHUTDOWN SYSTEM MONITORING INSTRUMENTATION

INSTRUMENT Function	READOUT[1] LOCATION	CHANNELS AVAILABLE	MINIMUM CHANNELS OPERABLE
1. Source Range Neutron Flux	A	1 (NI-31E)	1
2. Extended Range Neutron Flux	B	1 (NI-13135 C&D)	1
3. RCS LP Cold Leg Temperature	A, B	1/Loop (Loop 1 TI-0413D, Panel A) (Loop 2 TI-0423D, Panel B) (Loop 3 TI-0433D, Panel B) (Loop 4 TI-0443D, Panel A)	1/Loop
4. RCS Hot Leg Temperature	A	2 (Loop 1 TI-0413C Loop 4 TI-0443C)	2
5. Core Exit Thermocouples	B	2 (Loop 2 Core Quandrant TI-10055 Loop 3 Core Quandrant TI-10056)	2
6. RCS Wide Range Pressure	A, B	2 (PI-405A, Panel A) (PI-403A, Panel B)	2
7. Steam Generator Level Wide Range	A, B	1/Loop (Loop 1 LI-501B, Panel A) (Loop 2 LI-502B, Panel B) (Loop 3 LI-503B, Panel B) (Loop 4 LI-504B, Panel A)	1/Loop
8. Pressurizer Level	A, B	2 (LI-459C, Panel A) (LI-460C, Panel B)	2
9. RWST Level	L	1 (LI-0990C)	1[3]

TABLE 3.3-7 (Continued)

REMOTE SHUTDOWN SYSTEM MONITORING INSTRUMENTATION

INSTRUMENT Function	READOUT[1] LOCATION	CHANNELS AVAILABLE	MINIMUM CHANNELS OPERABLE
10. BAST Level	L	1 (PI-10115[2])	1[3]
11. CST Level	L	2 (Tank 1 LI-5100) (Tank 2 LI-5115)	2[3]
12. Auxiliary Feedwater Flow	A, B	1/LOOP (LOOP1 FI-5152B, Panel A) (LOOP2 FI-5151B, Panel B) (LOOP3 FI-5153B, Panel B) (LOOP4 FI-5150B, Panel A)	1/LOOP
13. Steam Generator Pressure	A, B	1/LOOP (LOOP1 PI-0514C, Panel A) (LOOP2 PI-0525B, Panel B) (LOOP3 PI-0535B, Panel B) (LOOP4 PI-0544C, Panel A)	1/LOOP

[1] A - Remote Shutdown Panel PSDA
B - Remote Shutdown Panel PSDB
L - Local Indication

[2] Graph will be provided to determine level from pressure reading

[3] Alternate local level indication may be established to fulfill the minimum channels OPERABLE.

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION: As shown in Table 3.3-8

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE:

a. Every 31 days by performance of a CHANNEL CHECK, and

b. Every 18 months by performance of a CHANNEL CALIBRATION.

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* The provisions of this specification are not applicable to the Unit 2 Containment Radiation Level (High Range) Monitors (Loop 0005 and 0006) until its initial entry into MODE 2.

TABLE 3.3-8

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Reactor Coolant Pressure (Wide Range) (Loop 408, 418, 428, & 438)	4	1	35
2. Reactor Coolant System T_{hot} (Wide Range) (Loop 413A, 423A, 433A & 443A)	1/loop	1/loop	32
3. Reactor Coolant System T_{cold} (Wide Range) (Loop 413B, 423B, 433B & 443B)	1/loop	1/loop	32
4. SG Water Level (Wide Range) (Loop 501, 502, 503 & 504)	1/SG	1/SG	32
5. SG Water Level (Narrow Range) (Loop 517, 518, 519, 527, 528, 529, 537, 538, 539, 547, 548, 549, 551, 552, 553, 554)	4/SG	1/SG	35
6. Pressurizer Level (Loop 459, 460, 461)	3	1	30
7. Containment Pressure (Loop 934, 935, 936, 937)	4	1	35
8. Steamline Pressure (Loop 514, 515, 516, 524, 525, 526, 534, 535, 536, 544, 545 & 546)	3/stm. line	1/stm. line	30
9. RWST Level (Loop 990, 991, 992 & 993)	4	1	35
10. Containment Normal Sumps Level (Narrow Range) (Loop 7777 & 7789)	2	1	31
11. Containment Water Level (Wide Range) (Loop 0764 & 0765)	2	1	31

TABLE 3.3-8 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
12. Condensate Storage Tank Level (Loop 5101, 5111, 5104 & 5116)	2/tank	1/tank	31
13. Auxiliary Feedwater Flow (Loop 5152, 15152, 5153, 15153, 5151, 15151, 5150 & 15150)	2/feed line	1/feed line	31
14. Containment Radiation Level (High Range) (Loop 0005 & 0006)	2	1	33
15. Steamline Radiation Monitor (Loop 13119, 13120, 13121 & 13122)	1/stm. line	1/stm. line	33
16. Core Exit Thermocouples	4/quad/train	2/quad/train	30
17. Reactor Coolant System Subcooling	2	1	31
18. Neutron Flux (Extended Range) (Loop 13135A & 13135B)	2	1	31
19. RVLIS	2	1	34
20. Containment Hydrogen Concentration (Loop 12979 & 12980)	2	1	31
21. Containment Pressure (Extended Range) (Loop 10942 & 10943)	2	1	31
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	36

*Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A or containment ventilation isolation signals).

TABLE 3.3-8 (Continued)

ACTION STATEMENTS

ACTION 30 - a. With the number of OPERABLE channels one less than the Total Number of Channels requirements, restore the inoperable channel to OPERABLE status within 31 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels two less then the Total Number of Channels requirement, restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE channels less than the Minimum Channels Operable requirement, restore at least one inoperable channel to OPERABLE status within 48 hours or be in HOT SHUTDOWN with the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

ACTION 31 - a. With the number of OPERABLE channels one less than the Total Number of Channels requirements, restore one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels less than the Minimum Channels Operable requirements, restore at least one inoperable channel to OPERABLE status within 48 hours, or be in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

ACTION 32 - With the number of OPERABLE channels less than the Minimum Channels Operable requirements, restore at least one inoperable channel to OPERABLE status within 48 hours, or be in HOT SHUTDOWN within the next 12 hours. The provisions of Specification 3.0.4 are not applicable.

ACTION 33 - With the number of OPERABLE channels less than the minimum channels OPERABLE requirement, initiate the alternate method of monitoring the parameter 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 6.8.2, within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status. The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3-8 (Continued)

ACTION STATEMENTS

ACTION 34 - With the number of OPERABLE channels less than the required number of channels or the minimum channels OPERABLE requirement, restore the inoperable channel(s) to OPERABLE status as per Action 31a or b as applicable if repair is feasible during plant operation. If repair is not feasible, prepare and submit a Special Report to the Commission, pursuant to Specification 6.8.2 within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status. The provisions of Specification 3.0.4 are not applicable.*

ACTION 35 - a. With the number of OPERABLE channels two less than the Total Number of Channels requirements, restore the inoperable channel to OPERABLE status within 31 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels three less than the Total Number of Channels requirement, restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE channels less than the Minimum Channels Operable requirement, restore at least one inoperable channel to OPERABLE status with 48 hours or be in HOT SHUTDOWN within the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

ACTION 36 - With the number of OPERABLE channels less than the minimum channels OPERABLE requirements, comply with the provisions of Specification 3.6.3. for an inoperable containment isolation valve.

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*Action Statement 34 applies to the first fuel cycle only. Action Statement 31 is applicable thereafter.

TO BE REVISED TO REFLECT UNIT 2

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent Chlorine Detection Systems (Loop 12110, Loop 12112) ~~(A-12110, A-12112)~~, with their Alarm/Trip Setpoints adjusted to actuate at a chlorine concentration of less than or equal to ~~2~~ 5 ppm, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5* and 6*, if chlorine gas is stored onsite in quantities greater than 20 lb.

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 Emergency Ventilation System in the isolation mode of operation.

b. With both Chlorine Detection Systems inoperable, within 1 hour initiate and maintain operation of the Control Room Emergency Ventilation System in the isolation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each Chlorine Detection System (Loop 12110, Loop 12112) ~~(A-12110, A-12112)~~ shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

*During movement of irradiated fuel or movement of loads over irradiated fuel.

LOOSE PARTS DETECTION SYSTEM

Specification 3/4 3.3.~~3~~.8 Deleted

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-9 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 and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

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 3.3-9. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.8.1.4 why this inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-5.

TABLE 3.3-9

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 (RE-0018)	1	37
b. Steam Generator Blowdown Effluent Line (RE-0021)	1	38
c. Turbine Building (Floor Drains) Sumps Effluent Line (RE-0848)	1	38
~~d. Control Building (Floor Drains) Sumps Effluent Line (RE-17646)~~	~~1~~	~~39~~
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
a. Nuclear Service Cooling Water System Effluent Line (RE-0020 A & B)	1	39
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line (FT-0018)	1	40
b. Steam Generator Blowdown Effluent Line (FT-0021)	1	40
c. Flow to Blowdown Sump (AFQI-7620, FR-7620, pen 1)	1	40

TABLE 3.3-9 (Continued)

ACTION STATEMENTS

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue 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 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microcurie/ml:

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

ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microcurie/ml.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

TABLE 4.3-5

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluent Line (RE-0018)	D	P	R(3)	Q(1)
b. Steam Generator Blowdown Effluent Line (RE-0021)	D	M	R(3)	Q(1)
c. Turbine Building (Floor Drains) Sumps Effluent Line (RE-0848)	D	M	R(3)	Q(1)
~~d. Control Building (Floor Drains) Sumps Effluent Line (RE-17646)~~	~~D~~	~~M~~	~~R(3)~~	~~Q(1)~~
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
Nuclear Service Cooling Water System Effluent Line (RE-0020 A & B)	D	M	R(3)	Q(2)

TABLE 4.3-5 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
3. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line (FT-0018)	D(4)	N.A.	R	N.A.
b. Steam Generator Blowdown Effluent Line (FT-0021)	D(4)	N.A.	R	N.A.
c. Flow to Blowdown Sump (AFQI-7620,FR-7620 pen 1)	D(4)	N.A.	R	Q

TABLE 4.3-5 (Continued)

TABLE NOTATIONS

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway (for item a. below only) and control room alarm annunciation occur if any of the following conditions exists:

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

b. Circuit failure, or

c. Instrument indicates a downscale failure, or

d. Instrument controls not set in operate mode. (annunciation via computer print-out)

(2) The ANALOG CHANNEL OPERATIONAL 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, or

d. Instrument controls not set in operate mode. (annunciation via computer print-out)

(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.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.1a and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1a shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-10

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 3.3-10. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.8.1.4 why this inoperability was not corrected in a timely manner.

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 ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-6.

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TABLE 3.3-10

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GASEOUS WASTE PROCESSING SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (ARE-0014)	1	***	45
b. Effluent System Flow Rate Measuring Device (AFT-0014)	1	***	46
2. GASEOUS WASTE PROCESSING SYSTEM Explosive Gas Monitoring System			
a. Hydrogen Monitor	1/recombiner	**	50
b. Oxygen Monitor	2/recombiner	**	49
3. Condenser Air Ejector and Steam Packing Exhauster System			
a. Noble Gas Activity Monitor (RE-12839C)	1	***	47
b. Iodine Sampler (RE-12839B)	1	***	51
c. Particulate Sampler (RE-12839A)	1	***	51
d. Flow Rate Monitor (FI-12839) (FIS-12862)#	1	***	46
e. Sampler Flow Rate Monitor (FI-13211)	1	***	46

TABLE 3.3-10 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. Plant Vent			
a. Noble Gas Activity Monitor (RE-12442C or RE-12444C)	1	*	47, 48
b. Iodine Sampler/Monitor (RE-12442B or RE-12444B)	1	*	51
c. Particulate Sampler/Monitor (RE-12442A or RE-12444A)	1	*	51
d. Flow Rate Monitor (FT-12442)	1	*	46
e. Sampler Flow Rate Monitor (FI-12442 or FI-12444)	1	*	46

TABLE 3.3-10 (Continued)

TABLE NOTATIONS

* At all times.
** During GASEOUS WASTE PROCESSING SYSTEM operation.
***During radioactive releases via this pathway.
\# During Emergency Filtration

ACTION STATEMENTS

ACTION 41-44 (Not Used)

ACTION 45 - 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 provided that prior to initiating the release:

a. At least two independent samples of the tank's 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 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 47 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 48 - 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 49 - a. With the outlet oxygen monitor channel inoperable, operation of the system may continue provided grab samples are taken and analyzed at least once per 24 hours and the oxygen concentration remains less than 1 percent.

b. With the inlet oxygen monitor inoperable, operation may continue if inlet hydrogen monitor is OPERABLE.

c. With both oxygen channels or both of the inlet oxygen and inlet hydrogen monitors inoperable, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations or at least once per 24 hours during other operations and the oxygen concentration remain less than 1 percent.

TABLE 3.3-10 (Continued)

TABLE NOTATIONS (Continued)

ACTION 50 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations or at least once per 24 hours during other operations and the oxygen concentration remains less than 1 percent.

ACTION 51 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

TABLE 4.3-6

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. GASEOUS WASTE PROCESSING SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (ARE-0014)	P	P	R(3)	Q(1)	c
b. Effluent System Flow Rate Measuring Device (AFT-0014)	P	N.A.	R	N.A.	c
2. GASEOUS WASTE PROCESSING SYSTEM Explosive Gas Monitoring System					
a. Hydrogen Monitors	D	N.A.	Q(4)	M	b
b. Oxygen Monitors	D	N.A.	Q(5)	M	b
3. Condenser Air Ejector and Steam Packing Exhauster System					
a. Noble Gas Activity Monitor (RE-12839C)	D	M	R(3)	Q(2)	c
b. Iodine Sampler (RE-12839B)	W(6)	N.A.	N.A.	N.A.	c
c. Particulate Sampler (RE-12839A)	W(6)	N.A.	N.A.	N.A.	c
d. Flow Rate Monitor (FT-12839)	D	N.A.	R	N.A.	c
e. Sampler Flow Rate Monitor (FI-13211)	D	N.A.	R	Q	c

TABLE 4.3-6 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Plant Vent					
a. Noble Gas Activity Monitor (RE-12442C or RE-12444C)	D	M	R(3)	Q(2)	a
b. Iodine Sampler (RE-12444B)	W(6)	N.A.	N.A.	N.A.	a
c. Particulate Sampler (RE-12444A)	W(6)	N.A.	N.A.	N.A.	a
d. Flow Rate Monitor (FT-12442)	D	N.A.	R	N.A.	a
e. Sampler Flow Rate Monitor (FI-12442 or FI-12444)	D	N.A.	R	Q	a
f. Particulate Monitor (RE-12442A)	D	N.A.	R	Q	a
g. Iodine Activity Monitor (RE-12442B)	D	N.A.	R	Q	a

TABLE 4.3-6 (Continued)

TABLE NOTATIONS

a At all times.
b During GASEOUS WASTE PROCESSING SYSTEM operation.
c During Radioactive Releases via this pathway.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway (for item a. below only) and control room alarm annunciation occurs if any of the following conditions exists:

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

b. Circuit failure, or

c. Instrument indicates a downscale failure, or

d. Instrument controls not set in operate mode. (annuciation via computer print-out)

(2) The ANALOG CHANNEL OPERATIONAL 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, or

d. Instrument controls not set in operate mode. (annuciation via computer print-out)

(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 in accordance with the manufacturer's recommendations. In addition, a standard gas sample of nominal four volume percent hydrogen, balance nitrogen, shall be used in the calibration to check linearity of the hydrogen analyzer.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples in accordance with the manufacturer's recommendations. In addition, a standard gas sample of nominal four volume percent oxygen, balance nitrogen, shall be used in the calibration to check linearity of the oxygen analyzer.

(6) The CHANNEL CHECK shall consist of visually verifying that the collection device (i.e., particulate filter or charcoal cartridge, etc.) is in place for sampling.

HIGH-ENERGY LINE BREAK ISOLATION SENSORS

LIMITING CONDITION FOR OPERATION

3.3.3.11 The high energy line break instrumentation listed in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: As noted in Table 3.3-11.

ACTION:

a. With the number of OPERABLE electric steam boiler isolation instruments less than the Minimum Channels OPERABLE as required by Table 3.3-11, restore the inoperable instruments to OPERABLE status within 7 days or suspend operation of the electric steam boiler until the inoperable sensors are restored to OPERABLE status. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE steam generator blowdow line isolation instruments or letdown line isolation instruments less than the Minimum Channels OPERABLE as required by Table 3.3.11, restore the inoperable instruments to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each of the above high energy line break isolation instruments shall be demonstrated OPERABLE by the performance of an ANALOG CHANNEL OPERATIONAL TEST at least once per 18 months.

TABLE 3.3-11

HIGH-ENERGY LINE BREAK INSTRUMENTATION

	Isolation Function	Instrument Channel	Minimum Channels Operable	Applicable Modes
1.	Electric Steam Boiler Isolation (Common Instrumentation)	A TE 19722A (RD52) A TE 19723A (RD52)	1	*
		A TE 19722B (RC41) A TE 19723B (RC41)	1	*
		A TE 19722C (RC65) A TE 19723C (RC64)	1	*
		A TE 19722D (RC67) A TE 19723D (RC66)	1	*
		A FT 19722 A FT 19723	1	*
		A TE 19722E (RC95) A TE 19723E (RC95)	1	*

	Isolation Function	Instrument Channel (Unit 1)	Instrument Channel (Unit 2)	Minimum Channels Operable	Applicable Modes
2.	Steam Generator Blowdown Line Isolation	TE 15212A (RB08) TE 15216A (RB08)	TE 15212A (RB131) TE 15216A (RB131)	1	1, 2, 3, 4
		TE 15212B (RC106) TE 15216B (RC106)	TE 15212B (RC 103) TE 15216B (RC 103)	1	1, 2, 3, 4
		TE 15212C (RC107) TE 15216C (RC107)	TE 15212C (RC 101) TE 15216C (RC 101)	1	1, 2, 3, 4
		TE 15212D (RC108) TE 15216D (RC108)	TE 15212D (RC 102) TE 15216D (RC 102)	1	1, 2, 3, 4
		FT 15212A (Loop 1) FT 15216A	FT 15212A (Loop 1) FT 15216A	1	1, 2, 3, 4
		FT 15212B (Loop 2) FT 15216B	FT 15212B (Loop 2) FT 15216B	1	1, 2, 3, 4
		FT 15212C (Loop 3) FT 15216C	FT 15212C (Loop 3) FT 15216C	1	1, 2, 3, 4
		FT 15212D (Loop 4) FT 15216D	FT 15212D (Loop 4) FT 15216D	1	1, 2, 3, 4
3.	Letdown Line Isolation	TE 15214A (A07) TE 15215A (A07)	TE 15214A (A100) TE 15215A (A100)	1	1, 2, 3, 4
		TE 15214B (A08) TE 15215B (A08)	TE 15214B (A101) TE 15215B (A101)	1	1, 2, 3, 4
		TE 15214C (A09) TE 15215C (A09)	TE 15214C (A103) TE 15215C (A103)	1	1, 2, 3, 4

*Required during all ~~modes~~ MODES when electric steam boiler is in operation.

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

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

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lines, or 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 isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. At least once per 7 days while in MODE 1 and while in MODE 2 with the turbine operating, 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) Six low pressure turbine intermediate stop valves, and

   3) Six low pressure turbine intercept valves.

b. At least once per 31 days while in MODE 1 and while in MODE 2 with the turbine operating, by direct observation of the movement of each of the above valves and the four high pressure turbine control 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 (including the four high pressure turbine control valves) and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

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*Not applicable in MODE 2 or 3 with all main steam line isolation valves and associated bypass valves in the closed position and all other steam flow paths to the turbine isolated.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION 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 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exceptions Specification 3.10.4.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with at least two reactor coolant loops in operation when the Reactor Trip System breakers are closed and at least 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 loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined 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 17% of wide range (LI-0501, LI-0502, LI-0503, LI-0504) at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant pumps may be deenergized 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.

**See Special Test Exception Specification 3.10.4.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops/trains listed below shall be OPERABLE and at least one of these loops/trains shall be in operation:*

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,**

d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,**

e. RHR train A, and

f. RHR train B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops/trains OPERABLE, immediately initiate corrective action to return the required loops/trains 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 loop/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 loop/train to operation.

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*All reactor coolant pumps and RHR pumps may be deenergized 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.

**A reactor coolant pump shall not be started unless the secondary water temperature of each steam generator is less than 50 °F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined 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 water level to be greater than or equal to 17% of wide range (LI-0501, LI-0502, LI-0503, LI-0504) at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR train shall be verified in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION 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 17% of wide range (LI-0501, LI-0502, LI-0503, LI-0504).

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR trains inoperable or with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR train to OPERABLE status or restore the required steam generator water 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 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.

4.4.1.4.1.2 At least one RHR train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

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*The RHR pump may be deenergized 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.

**One RHR train may be inoperable for up to 2 hours for surveillance testing provided the other RHR train is OPERABLE and in operation.

***A reactor coolant pump shall not be started unless the secondary water temperature of each steam generator is less than 50 °F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION 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.** Reactor Makeup Water Storage Tank (RMWST) discharge valves (1208-U4-175, 1208-U4-176, 1208-U4-177 and 1208-U4-183) shall be closed and secured in position.

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR trains OPERABLE, immediately initiate corrective action to return the required RHR 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.

c. With the Reactor Makeup Water Storage Tank (RMWST) discharge valves (1208-U4-175, 1208-U4-176, 1208-U4-177, and 1208-U4-183) not closed and secured in position, immediately close and secure in position the RMWST discharge valves.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 At least one RHR train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.4.2.2 Valves 1208-U4-175, 1208-U4-176, 1208-U4-177, and 1208-U4-183 shall be verified closed and secured in position by mechanical stops at least once per 31 days.

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*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 deenergized 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.

3/4.4.2 SAFETY VALVES

SHUTDOWN

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 an OPERABLE RHR 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.

*While in MODE 5, an equivalent size vent pathway may be used provided that the vent pathway is not isolated or sealed.

**The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION 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:

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.

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 CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 92% (1656 cubic feet), (LI-0459A, LI-0460A, LI-0461A) and at least two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, 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 System 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 energizing the heaters and measuring circuit current at least once per 92 days.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION 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, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, 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, and

   1. With only one PORV OPERABLE, restore at least a total of two PORVs to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

   2. With no PORVs OPERABLE, restore at least one PORV to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

c. With one or more block valve(s) inoperable, within 1 hour (1) restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s) or close the PORV and remove power from its associated solenoid valve; and (2) apply ACTION b above, as appropriate, for the isolated PORV(s).

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 Each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the valve through one complete cycle of full travel, and

b. Performing a CHANNEL CALIBRATION.

* The provisions of this specification are not applicable to Unit 2 until initial entry of Unit 2 into MODE 2.

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

SURVEILLANCE REQUIREMENTS

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 in order to meet the requirements of ACTION a and b in Specification 3.4.4.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION 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} (TI-0413A&B, TI-0433A&B, TI-0423A&B, TI-0443A&B) 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.

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%,

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.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.

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 .ffective 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) Primary-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 Operating Basis 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.

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 an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c.,

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 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.

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 6.8.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.8.2 within 12 months following the 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 shall be reported in a special report to the Commission pursuant to specification 6.8.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

TABLE NOTATIONS

[1]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 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.

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. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR 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. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR 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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System,

b. The Containment Normal Sumps Level and Reactor Cavity Sump Level, and

c. Either the containment air cooler condensate flow rate or a Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System not taken credit for in 3.4.6.1a.

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 or Particulate Radioactive 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 Gaseous and Particulate Monitoring Systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment Normal Sumps Level and Reactor Cavity Sump Level performance of CHANNEL CALIBRATION at least once per 18 months, and

c. Containment Air Cooler Condensate Monitoring System performance of CHANNEL CALIBRATION at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION 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 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. 64 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. 0.5 GPM leakage per nominal inch of valve size up to a maximum of 5 GPM at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve is 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 limit specified in Table 3.4-1, 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 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.

---

*Test pressures less than 2235 psig but greater than 350 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

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 (gaseous or particulate) radioactivity monitor at least once per 12 hours;

b. Monitoring the containment normal sumps and reactor cavity sump inventory at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days. 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; 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 by verifying leakage to be within its limit:

a. At least once during each refueling outage (leak testing should be performed after all disturbances to the valves are complete, such as before reaching power operation during a refueling outage);

b. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve (leak testing should be performed after all disturbances to the valves are complete);

c. For systems rated at less than 50% of RCS design pressure, each time the valve has moved from its fully closed position except for valves HV-8701 A/B and HV-8702 A/B.

d. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months except for valves HV-8701 A/B and HV-8702 A/B.

e. As outlined in the ASME Code, Section XI, paragraph IWV-3427(b).

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	VALVE SIZE(in.)	FUNCTION	MAXIMUM ALLOWABLE LEAKAGE(gpm)
1.	HV-8701A	12	RHR Suction (gate valve)	5.0
2.	HV-8701B	12	RHR Suction (gate valve)	5.0
3.	HV-8702A	12	RHR Suction (gate valve)	5.0
4.	HV-8702B	12	RHR Suction (gate valve)	5.0
5.	1204-U4-120	2	SI-Hot Leg 2nd Isolation Valve	1.0
6.	1204-U4-121	2	SI-Hot Leg 2nd Isolation Valve	1.0
7.	1204-U4-122	2	SI-Hot Leg 2nd Isolation Valve	1.0
8.	1204-U4-123	2	SI-Hot Leg 2nd Isolation Valve	1.0
9.	1204-U6-079	10	Accumulator 2nd Isolation Valve	5.0
10.	1204-U6-080	10	Accumulator 2nd Isolation Valve	5.0
11.	1204-U6-081	10	Accumulator 2nd Isolation Valve	5.0
12.	1204-U6-082	10	Accumulator 2nd Isolation Valve	5.0
13.	1204-U6-083	10	Injection Line 1st Isolation Valve	5.0
14.	1204-U6-084	10	Injection Line 1st Isolation Valve	5.0
15.	1204-U6-085	10	Injection Line 1st Isolation Valve	5.0
16.	1204-U6-086	10	Injection Line 1st Isolation Valve	5.0
17.	1204-U6-124	6	SI-Hot Leg 1st Isolation Valve	3.0
18.	1204-U6-125	6	SI-Hot Leg 1st Isolation Valve	3.0
19.	1204-U6-126	6	SI-Hot Leg 1st Isolation Valve	3.0
20.	1204-U6-127	6	SI-Hot Leg 1st Isolation Valve	3.0
21.	1204-U6-128	8	RHR-Hot Leg 2nd Isolation Valve	4.0
22.	1204-U6-129	8	RHR-Hot Leg 2nd Isolation Valve	4.0
23.	1204-U4-143	2	SI-Cold Leg 2nd Isolation Valve	1.0
24.	1204-U4-144	2	SI-Cold Leg 2nd Isolation Valve	1.0
25.	1204-U4-145	2	SI-Cold Leg 2nd Isolation Valve	1.0
26.	1204-U4-146	2	SI-Cold Leg 2nd Isolation Valve	1.0
27.	1204-U6-147	6	RHR Cold Leg 2nd Isolation Valve	3.0
28.	1204-U6-148	6	RHR Cold Leg 2nd Isolation Valve	3.0
29.	1204-U6-149	6	RHR Cold Leg 2nd Isolation Valve	3.0
30.	1204-U6-150	6	RHR Cold Leg 2nd Isolation Valve	3.0

3/4.4.7 CHEMISTRY

LIMITING CONDITION 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 specified in Table 3.4-2 at least once per 72 hours.*

*Sample and analysis for dissolved oxygen is not required with $T_{avg} \leq 250°F$.

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} less than or equal to 250°F.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per 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

b. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per 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:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram of gross radioactivity, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

*With T_{avg} greater than or equal to 500°F.

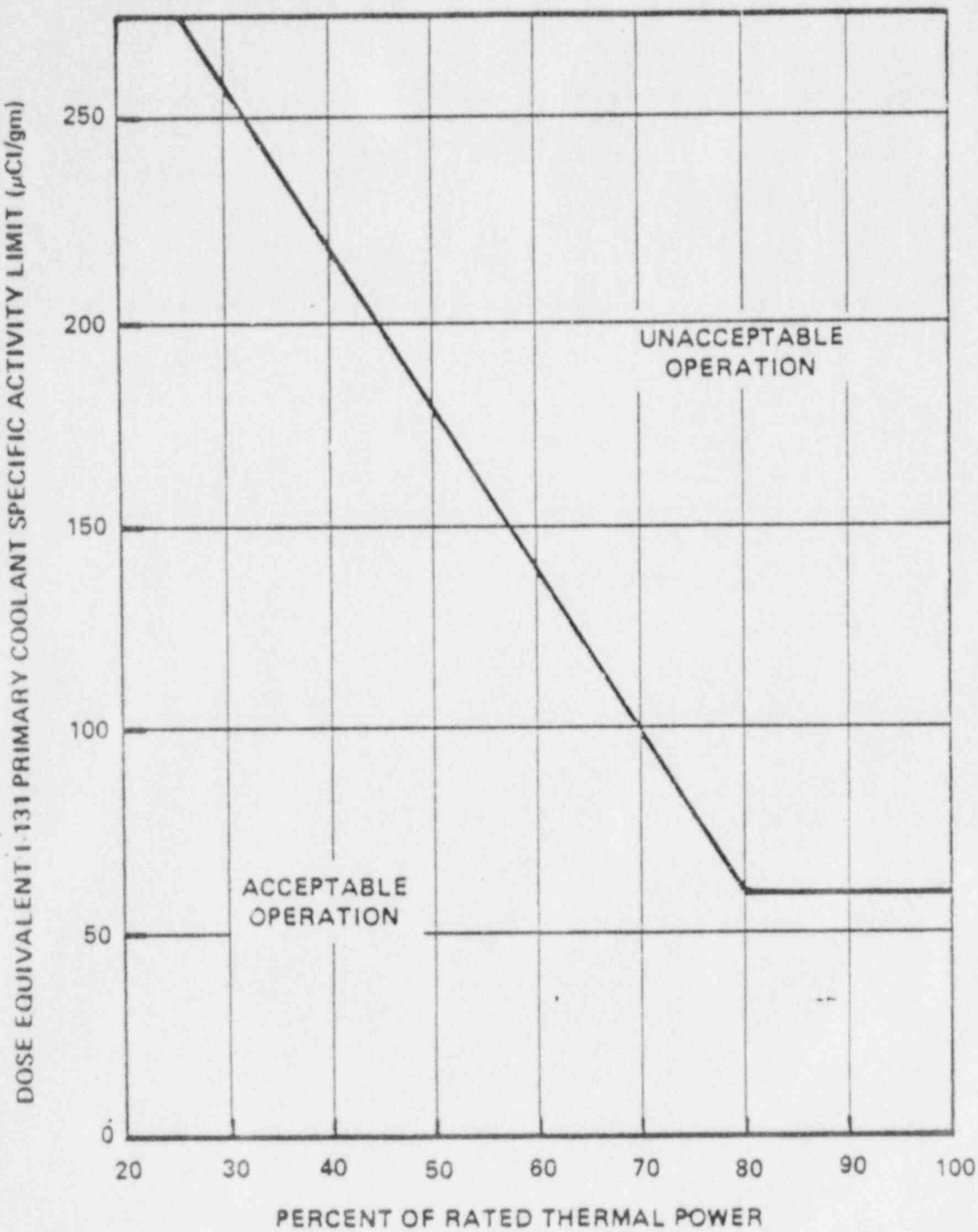


FIGURE 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 µCi/gram DOSE EQUIVALENT I-131

TABLE 4.4-3

NOT USED

TABLE 4.4-4

REACTOR 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 Radioactivity 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 Analysis for $\bar{E}$ 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 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

TABLE NOTATIONS

*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 14 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 data may be used for pure beta-emitting radionuclides.

**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 14 minutes and all radioiodines, which are 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 those energy peaks identifiable with a 95% confidence level.

***Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the Reactor Coolant System is restored within its limits.

TO BE REVISED TO REFLECT UNIT 2

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION 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 1-hour period,

b. A maximum cooldown of 100°F in any 1-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.

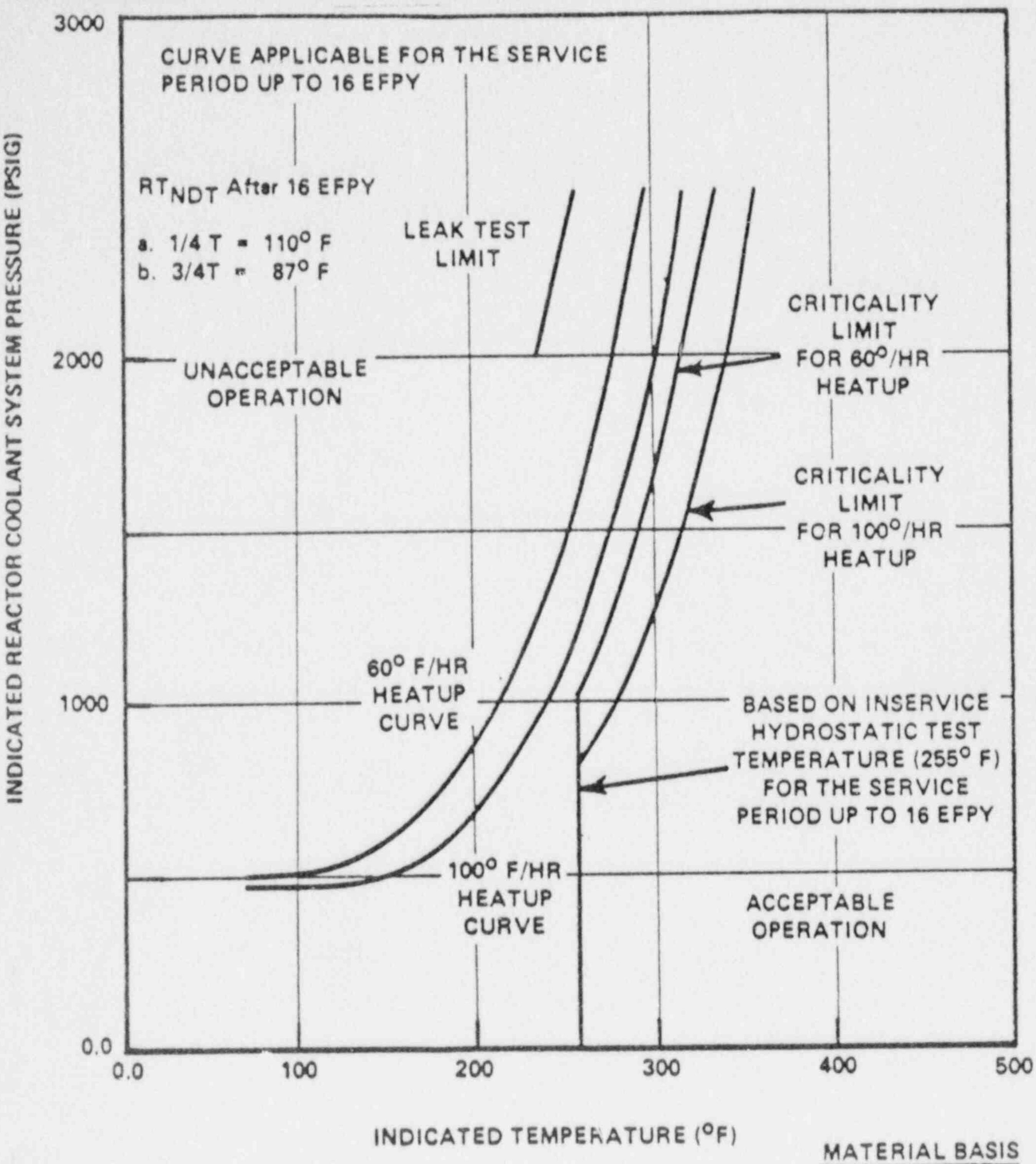
APPLICABILITY: At all times.

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 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.



MATERIAL BASIS

Copper Content Assumed – .10 Wt %
(Actual – .06 Wt %

RT_{NDT} Initial Assumed – 40° F
(Actual – 30° F)

RT_{NDT} After 16 EFPY @ 1/4T · 110° F
@ 3/4T · 87° F

TO BE REVISED TO REFLECT UNIT 2

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 16 EFPY

TO BE REVISED TO REFLECT UNIT 2

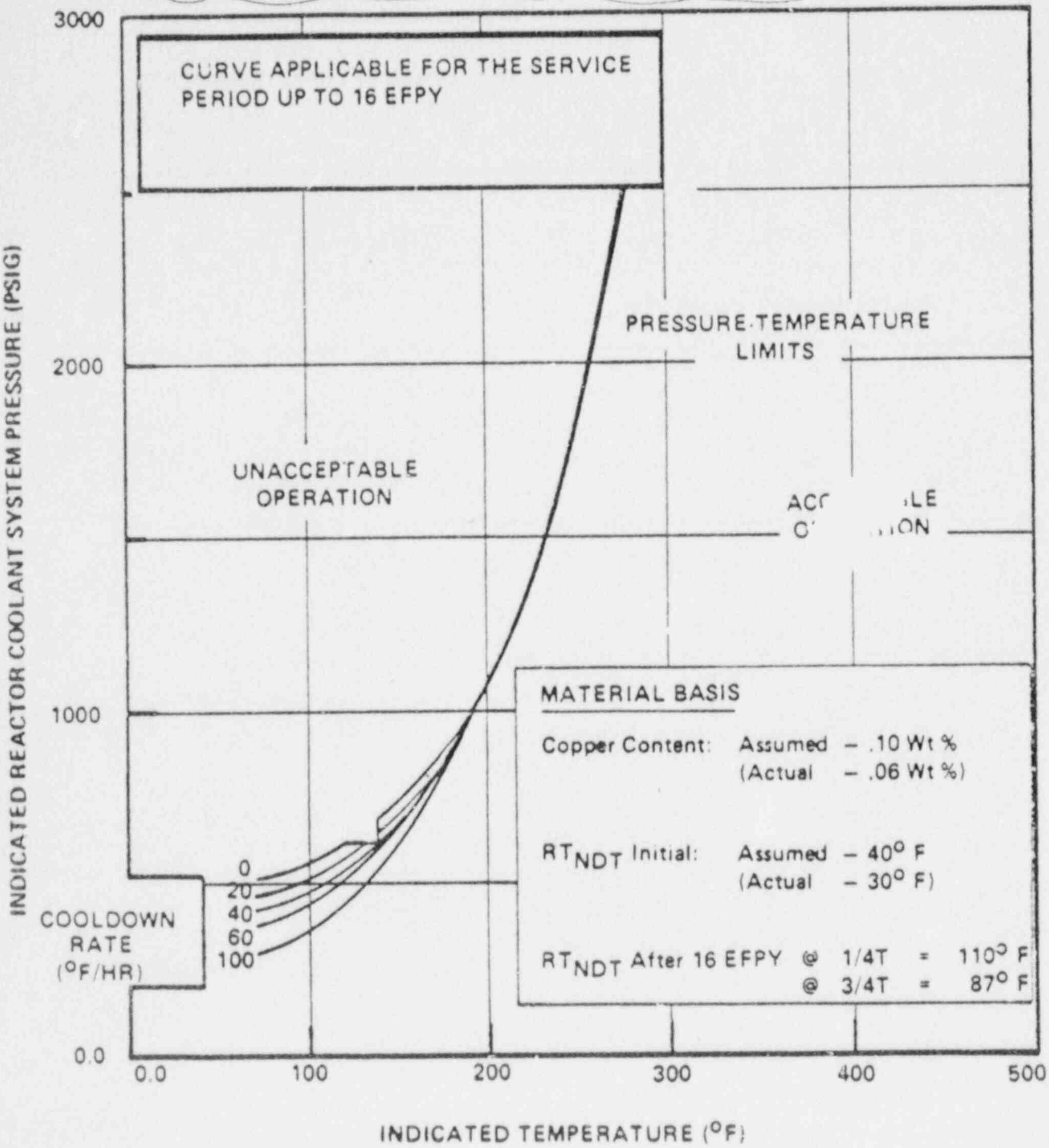


FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 16 EFPY

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature (TI-0453, TI-0454) shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum auxiliary spray water temperature differential of 625°F (TI-0126).

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 30 minutes 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.

COLD OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Cold Overpressure Protection Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with lift settings which vary with RCS temperature and which do not exceed the limits established in Figure 3.4-4, or

b. Two residual heat removal (RHR) suction relief valves each with a setpoint of 450 psig ± 3%, or

c. The Reactor Coolant System (RCS) depressurized with an RCS vent capable of relieving at least 670 GPM water flow at 470 psig.

APPLICABILITY: MODES 4, 5, and 6 with the reactor vessel head on.

ACTION:

a. With one PORV and one RHR suction relief valve inoperable, either restore two PORVs or two RHR suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS as specified in 3.4.9.3.c, above, within the next 8 hours.

b. With both PORVs and both RHR suction relief valves inoperable, depressurize and vent the RCS as specified in 3.4.9.3.c, above, within 8 hours.

c. In the event either the PORVs, the RHR suction relief valves, 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 6.8.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, the RHR suction relief valves or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

TO BE REVISED TO REFLECT UNIT 2

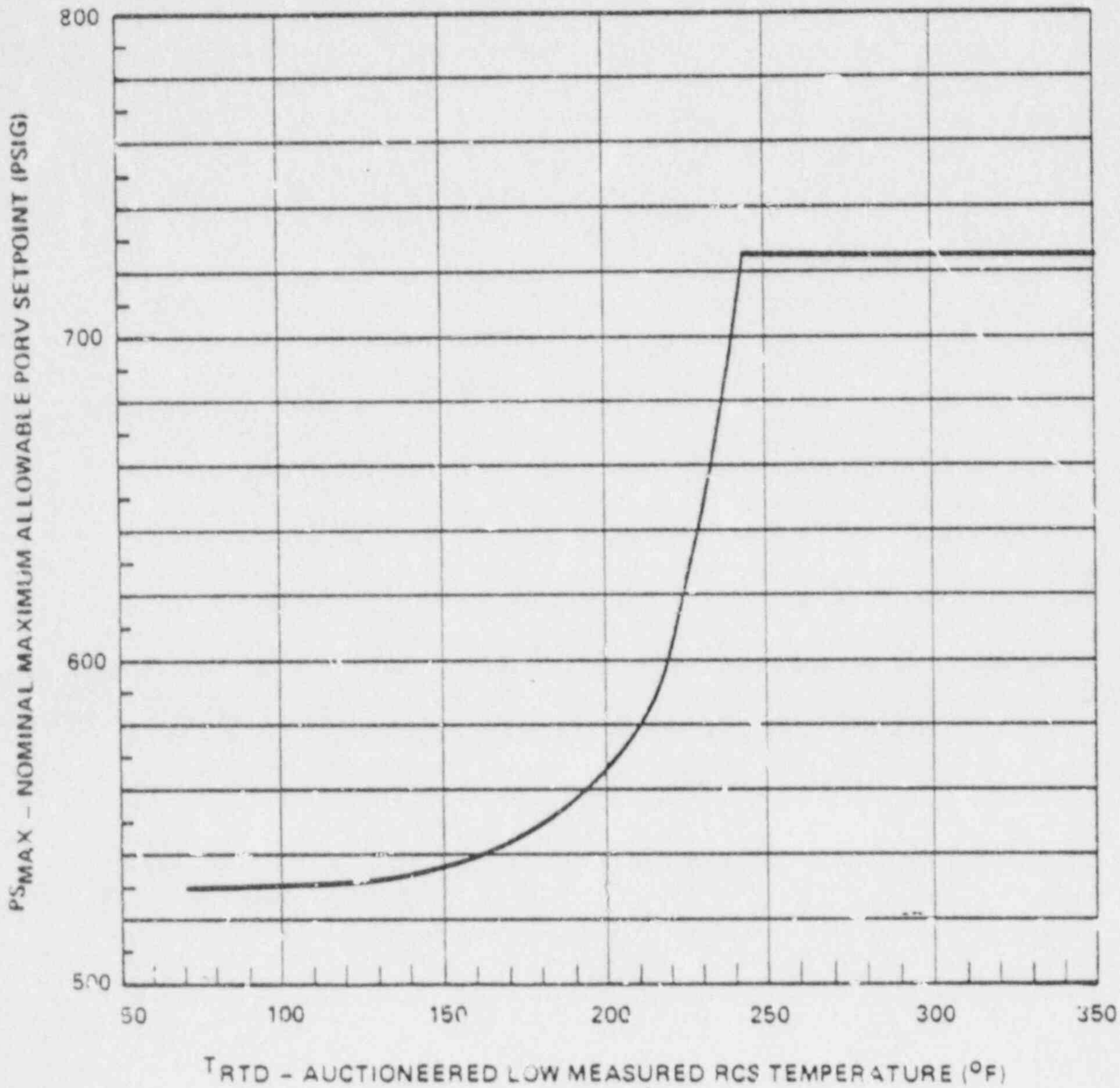


FIGURE 3.4-4

MAXIMUM ALLOWABLE NOMINAL PORV SETPOINT
FOR THE COLD OVERPRESSURE PROTECTION SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL 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 and at least once per 31 days thereafter when 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.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve PSV-8708A by verifying at least once per 12 hours that RHR RCS suction isolation valves HV-8701A and HV-8701B are open;

b. For RHR suction relief valve PSV-8708B by verifying at least once per 12 hours that RHR RCS suction isolation valves HV-8702A and HV-8702B are open; and

c. Testing pursuant to specification 4.0.5.

4.4.9.3.3 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.

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION 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

4.4.10.1 Not Used

4.4.10.2 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.

4.4.10.3 In addition to the requirements of Specification 4.0.5, the four main steam lines and feedwater lines from the containment penetration flued head outboard welds, up to the first five-way restraint, shall be inspected. The extent of the inservice examinations completed during each inspection interval (ASME Code Section XI) shall provide 100 percent volumetric examination of circumferential and longitudinal welds to the extent practical.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 Two reactor vessel head vent paths each consisting of two vent valves and a control valve powered from emergency busses, shall be OPERABLE and closed.

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

ACTION:

a. With one of the above reactor vessel head vent paths 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 vent 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.

b. With two reactor vessel head vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuators of all the vent valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each Reactor Coolant System 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 vent valve and control valve through at least one complete cycle of full travel from the control room, and

c. Verifying flow through the Reactor Coolant System vent paths during venting.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 6616 (36% of instrument span) and 6854 (64% of instrument span) gallons (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

c. A boron concentration of between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between 617 and 678 psig. (PI-0960A&B, PI-0961A&B, PI-0962A&B, PI-0963A&B, PI-0964A&B, PI-0965A&B, PI-0966A&B PI-0967A&B)

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 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig 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 at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the 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 [illegible] tanks, and

2) Verifying that each accumulator isolation valve is open. (HV-8808A, B, C, D)

---

*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 (67 gallons) by verifying the boron concentration of the accumulator solution; and

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the circuit breaker supplying power to the isolation valve operator is open.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent 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 RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and semi-automatically transferring suction to the containment emergency 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 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 6.8.2 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.

MODE

* The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into ~~Mode~~ 3 for the Safety Injection Pumps declared inoperable pursuant to Specification 3.5.3.2 provided the Safety Injection Pumps are restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever occurs first.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power lockout switches in the lockout position:

Valve Number	Valve Function	Valve Position
HV-8835	SI Pump Cold Leg. Inj.	OPEN
HV-8840	RHR Pump Hot Leg. Inj.	CLOSED
HV-8813	SI Pump Mini. Flow Isol.	OPEN
HV-8806	SI Pump Suction from RWST	OPEN
HV-8802A, B	SI Pump Hot Leg Inj.	CLOSED
HV-8809A, B	RHR Pump Cold Leg Inj.	OPEN

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.

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 Emergency 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:

1) Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 377 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 750 psig the interlocks will cause the valves to automatically close.

2) A visual inspection of the Containment Emergency 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 abnormal corrosion.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months, during shutdown, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal,

   2) Verifying that, for the RHR Semi-Automatic Switchover to Containment Emergency Sump, each automatic valve in the flow path actuates to its correct position on RWST level low-low coincident with Safety Injection test signal, and

   3) 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) RHR 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 $\geq$ 2364 psid,

   2) Safety Injection pump $\geq$ 1405 psid, and

   3) RHR pump $\geq$ 165 psid.

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.

Safety Injection System Valve Number	
1204-U4-022	1204-U4-139
1204-U4-023	1204-U4-140
1204-U4-024	1204-U4-141
1204-U4-025	1204-U4-142
1204-U4-116	1204-U4-118
1204-U4-117	1204-U4-119

SURVEILLANCE REQUIREMENTS (Continued)

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 284 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 406 gpm, and

b) The total pump flow rate is less than or equal to 660 gpm.

3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3788 gpm.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

ECCS SUBSYSTEMS

LIMITING CONDITION FOR OPERATION

3.5.3.1 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR 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 Emergency 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 removal heat exchanger or RHR 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 6.8.2 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

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

SAFETY INJECTION PUMPS

LIMITING CONDITION FOR OPERATION

3.5.3.2 All Safety Injection pumps shall be inoperable.

APPLICABILITY MODES 4, 5, and 6 with the reactor vessel head on.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.3.2 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, and at least once per 31 days thereafter.

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* An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 631,478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A).

b. A boron concentration of between 2000 ppm and 2100 ppm of boron,

c. A minimum solution temperature of 54°F, and

d. A maximum solution temperature of 116°F (TI-10982).

e. RWST Sludge Mixing Pump Isolation valves capable of closing on RWST low-level.

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

ACTION:

a. With the RWST inoperable except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed..

SURVEILLANCE REQUIREMENTS

4.5.4 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 air temperature is less than 50°F.

c. At least once per 18 months by verifying that the sludge mixing pump isolation valves automatically close upon an RWST low-level test signal.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION 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 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;

b. By verifying that each containment air lock is in compliance with the requirements 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 a pressure not less than P_a, 45 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 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 that such verification need not be performed more often than once per 92 days. The blind flange on the fuel transfer canal need not be verified closed except after each draining of the canal.

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 less than or equal to L_a, 0.20% by weight of the containment air per 24 hours at a pressure not less than P_a, 45 psig, or

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 a pressure not less than P_a, 45 psig.

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

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a or 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 0.75 L_a, 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.

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 Part 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 a pressure not less than P_a, 45 psig, 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;

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet 0.75 L_a 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 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a 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 test by verifying that the absolute value of the supplemental test result, 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;

   2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

   3) 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.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 45 psig, at intervals no greater than 24 months except for tests involving:

   1) Air locks,

   2) Purge supply and exhaust isolation valves with resilient material seals,

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.2;

g. 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 P_a, 45 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 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 6 hours and in COLD SHUTDOWN within the following 30 hours.

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*Momentary entry to repair the inner air lock door, if inoperable, is permissible for a period not to exceed 15 minutes per repair.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage is less than 0.01 L_a as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of 45 psig;

b. By conducting overall air lock leakage tests at not less than P_a, 45 psig, and verifying the overall air lock leakage rate is within its limit:

   1) At least once per 6 months,* and

   2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.**

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

*The provisions of Specification 4.0.2 are not applicable.

**This represents an exemption to Appendix J, paragraph III.D.2 of 10 CFR Part 50.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure (PI-0934, PI-0935, PI-0936, PI-0937, P-9871) shall be maintained between -0.3 and + 1.8 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 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

4.6.1.4 The primary containment internal pressure (PI-0934, PI-0935, PI-0936, PI-0937, P-9871) shall be determined to be within the limits at least once per 12 hours.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature (TE-2563, TE-2612, TE-2613) 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:

Location	Tag Numbers*
a. Level 2	TE-2563
b. Level B	TE-2613
c. Level C	TE-2612

*Or local sample at corresponding location

Insert to Page 3/4 6-8.

Insert 1 In addition, within 90 days of completion of the Unit 1 evaluation, perform an engineering evaluation of Unit 2 containment and provide a Special Report to the Commission in accordance with Specification 6.8.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Insert 2 b. With any abnormal degradation of the Unit 1 structural integrity as defined in 4.6.1.6.1.1 items b or e, restore the containment to the required level of integrity within 72 hours and perform an engineering evaluation of Unit 1 containment and provide a Special Report to the Commission with 15 days in accordance with Specification 6.8.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. In addition, within 90 days of completion of the Unit 1 evaluation, perform an engineering evaluation of the Unit 2 containment and provide a Special Report to the Commission in accordance with Specification 6.8.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of ~~the~~ each 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:

a. With more than one Unit 1 tendon with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the Unit 1 containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.8.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. INSERT 1

INSERT 2 HERE

c. ~~b.~~ With any abnormal degradation of the structural integrity other than ACTIONS a. and b. at a level below the acceptance criteria of Specification 4.6.1.6, restore the applicable containment to the required level of integrity within 72 hours and perform an engineering evaluation of the applicable containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.8.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1.1 Containment Tendons. The Unit 1 containment tendons' structural integrity of the Unit 1 shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. The ~~tendons'~~ structural integrity of the tendons shall be demonstrated by:

a. Determining that a random but representative sample of at least 13 tendons (4 inverted U and 9 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and acceptable. Unless there is abnormal degradation of the containment during the first three inspections, the sample population for subsequent inspections shall include at

SURVEILLANCE REQUIREMENTS (Continued)

least 5 tendons (2 inverted U and 3 hoop). This test shall include essentially complete detensioning of each tendon to determine if any strands are broken or damaged.

b. Removing a strand from one hoop tendon and one inverted U tendon and performing an inspection and material test. It shall be determined that over the entire length of both removed strands:

   1) The tendon wires or strands are free of corrosion, cracks, and damage,

   2) There are not changes in the presence or physical appearance of the sheathing filler-grease, and

   3) A minimum tensile strength of 270,000 psi (guaranteed ultimate strength of the tendon material) for at least three strand samples (one from each end and one at mid-length) cut from each removed strand. Failure of any one of the strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of +6%. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the-difference is not related to wire failures or slip of wires in anchorages;

d. Assuring the observed lift-off forces are between the maximum and minimum values given in the tendon surveillance procedure.

e. Verifying the OPERABILITY of the sheathing filler grease by:

   1) No voids in excess of 5% of the net duct volume,

   2) Minimum grease coverage exists for the different parts of the anchorage system, and

   3) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

4.6.1.6.3 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.

4.6.1.6.1.2 Containment Tendons - Unit 2. The structural integrity of the Unit 2 containment tendons shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. The structural integrity of the tendons shall be demonstrated by a visual examination of the tendon anchorage hardware (to the extent practical and without dismantling load bearing components of the anchorage) of a representative sample of tendons selected in the same manner as described in Specification 4.6.1.6.1.1.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve (HV-2626A&B, HV-2627A&B, HV-2628A&B, HV-2629A&B) shall be OPERABLE and:

a. Each 24-inch containment purge supply and exhaust isolation valve shall be closed and sealed closed, and

b. The 14-inch containment purge supply and exhaust isolation valve(s) shall be closed to the maximum extent practicable but may be open for purge system operation for pressure control, for ALARA and respirable air quality considerations for personnel entry and for surveillance tests that require the valve(s) to be open.

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

ACTION:

a. With a 24-inch containment purge supply and/or exhaust isolation valve open or not sealed closed, close and seal that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the 14-inch containment purge supply and/or exhaust isolation valve(s) open for reasons other than given in 3.6.1.7b above, close the open 14-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specification 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status within 24 hours, 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 Each 24-inch containment purge supply and exhaust isolation valve (HV-2626A, HV-2627A, HV-2628A, HV-2629A) shall be verified to be sealed closed at least once per 31 days.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.7.2 At least once per 3 months the containment purge valves with resilient material seals in each sealed closed containment purge supply and exhaust penetration shall be demonstrated OPERABLE by verifying that the measured penetration leakage rate is less than 0.06 La when pressurized to Pa.

4.6.1.7.3 Each 14-inch containment purge supply and exhaust isolation valve (HV-2626B, HV-2627B, HV-2628B, HV-2629B) shall be verified to be closed or open in accordance with Specification 3.6.1.7b at least once per 31 days.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment emergency 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 196 psid when tested pursuant to Specification 4.0.5;

c. At least once per 18 months during shutdown, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation test signal, and

   2) Verifying that each spray pump starts automatically on a containment spray actuation 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 containing a volume of between 3700 89.9% ~~(87.7%)~~ and 4000 ~~(97.4%)~~ 97.2% gallons (LI-0931A, LI-0931B) of between 30 and 32% by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the spray 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 during shutdown, 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 each eductor suction flow rate (to be determined during preoperational tests) by isolating the spray additive tank, opening the valves in the miniflow lines, and the valve in the eductor test line, and running the respective pump:

1) Train A 130 ± 30 gpm (Unit 1), and ______ ± ______ gpm (Unit 2).

2) Train B 120 ± 30 gpm (Unit 1), and ______ ± ______ gpm (Unit 2).

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Two independent trains of containment cooling fans shall be OPERABLE with four fans to each train.

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

ACTION:

With one train of the above required containment cooling fans inoperable, restore the inoperable train of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable train of cooling fans to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each train of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Starting each fan pair from the control room, and verifying that each fan pair operates at low speed for at least 15 minutes, and

2) Verifying a cooling water flow rate of greater than or equal to 1359 gpm per pair of containment fan coolers (FR-1818/1&2, FR-1819/1&2).

b. At least once per 18 months by verifying that each fan train starts automatically and operates at low speed on a Safety Injection test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE.

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

ACTION:

With one or more of the containment isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

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.

e. The provisions of Specification 3.0.4 are not applicable provided that the affected penetration is isolated in accordance with ACTION b or c above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The containment isolation valves 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.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Containment Ventilation Isolation test signal, each containment ventilation isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors (AI-12979, AI-12980) shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen monitors inoperable, restore at least one 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 monitor (AI-12979, AI-12980) shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent 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 Hydrogen Recombiner System functional test, that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 kW, and

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 four reactor coolant loops and associated steam generators in operation and 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 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. 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	65
3	43

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER					LIFT SETTING (± 1%)*	ORIFICE SIZE
	SG-1	SG-2	SG-3	SG-4		
1.	PSV 3001	PSV 3011	PSV 3021	PSV 3031	1185 psig	16.0 in^2
2.	PSV 3002	PSV 3012	PSV 3022	PSV 3032	1200 psig	16.0 in^2
3.	PSV 3003	PSV 3013	PSV 3023	PSV 3033	1210 psig	16.0 in^2
4.	PSV 3004	PSV 3014	PSV 3024	PSV 3034	1220 psig	16.0 in^2
5.	PSV 3005	PSV 3015	PSV 3025	PSV 3035	1235 psig	16.0 in^2

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent 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 emergency 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 31 days on a STAGGERED TEST BASIS by:

1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1605 psig at a flow of greater than or equal to ~~175~~ 150 gpm (FI-15101, FI-15102);

2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1675 psig at a flow of greater than or equal to 145 gpm (FI-~~5100~~ 15100) when the secondary steam supply pressure (PI-5105A, PI-5105B) is greater than 900 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

* The provisions of this specification are not applicable to Unit 2 until its initial entry into MODE 2.

SURVEILLANCE REQUIREMENTS (Continued)

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; and

4) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is in standby for auxiliary feedwater automatic initiation or when above 10% RATED THERMAL POWER.

b. At least once per 18 months during shutdown by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 Condensate Storage Tank (CST) V4001 shall be OPERABLE with a contained water volume of at least 340,000 gallons (56% of instrument span) of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With CST V4001 inoperable, within 4 hours either:

a. Restore CST V4001 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 CST V4002 and restore CST V4001 to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 CST V4001 shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume (LI-5101, LI-5111A) is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 CST V4002 shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume (LI-5104, LI-5116A) ~~is~~ at least 340,000 gallons when it is the supply source for the auxiliary feedwater pumps.

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 0.1 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 0.1 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 Radioactivity Determination*	At least once per 72 hours.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) Once per 6 months, whenever the gross radioactivity determination indicates 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 14 minutes. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Two main steam line isolation systems (each system consisting of a main steam isolation valve (MSIV) and its associated bypass valve (MSIBV)) per steam line shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

a. With two main steam line isolation systems in any one steam line inoperable; POWER OPERATION may continue provided each MSIV in the affected steam line is open and at least one main steam line isolation system in the affected steam line is restored to OPERABLE status within 4 hours; otherwise reduce power to less than or equal to 5% of RATED THERMAL POWER within 2 hours.

b. With one main steam line isolation system inoperable, power operation may continue provided the MSIV in the affected isolation system is open and the inoperable system is restored to OPERABLE status within 3 days. Otherwise, reduce power to less than or equal to 5% of RATED THERMAL POWER within 2 hours.

MODES 2 and 3:

a. With two main steam line isolation systems in any one steam line inoperable, subsequent operation in MODES 2 or 3 may proceed provided at least one main steam line isolation system in the affected steam line is maintained closed. The provisions of Specification 3.0.4 are not applicable. Otherwise, be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b. With one main steam line isolation system inoperable but closed, subsequent operation in MODES 2 or 3 may proceed provided that the isolation system in the affected steam line is maintained closed. The provisions of Specification 3.0.4 are not applicable.

c. With one main steam line isolation system inoperable but open, either close the OPERABLE isolation system or restore the inoperable system to OPERABLE status within 7 days. Otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV and MSIBV 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.

*An OPERABLE main steam line isolation system can consist of an OPERABLE MSIV and an inoperable associated MSIBV provided the inoperable MSIBV is maintained closed.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor (TI-0413B, TI-0423B, TI-0433B, TI-0443B) and secondary (TI-1175, TI-1176, TI-1177, TI-1178) coolants in the steam generators shall be greater than 70°F when the pressure, (PI-0408, PI-0418, PI-0428, PI-0438, PI-0514A, PI-0524A, PI-0534A, PI-0544A) 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 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 reactor (TI-0413B, TI-0423B, TI-0433B, TI-0443B) or secondary (TI-1175, TI-1176, TI-1177, TI-1178) coolant is less than 70°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 Two independent component cooling water trains shall be OPERABLE with at least two pumps per train.

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

ACTION:

With only one component cooling water train OPERABLE, restore the inoperable train 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 At least two component cooling water trains shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that each Component Cooling Water System pump starts automatically on a Safety Injection test signal.

3/4.7.4 NUCLEAR SERVICE COOLING WATER (NSCW) SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 Two independent NSCW trains shall be OPERABLE with at least two pumps per train.

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

ACTION:

With only one NSCW train OPERABLE, restore the inoperable train 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 At least two NSCW trains 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 during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection test signal, and

2) Each NSCW ~~Service Water System~~ pump starts automatically on a Safety Injection test signal.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The Ultimate Heat Sink (UHS) shall be OPERABLE with:

a. Two OPERABLE Nuclear Service Cooling Water (NSCW) tower basins each with:

1. A minimum water level (LI-1606-train A, LI-1607-train B) in the NSCW tower basin of 80.25 ft (plant elevation of 217' 3") (73% of instrument span)

2. A maximum water temperature (TJR-1690/1-train A, TJR-1691/1-train B) of 90°F.

b. Two OPERABLE trains of NSCW tower fans, each train consisting of four fans and associated spray cells.

c. Two OPERABLE NSCW transfer pumps.

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

ACTION:

a. With the UHS inoperable due to water level and/or water temperature, restore the UHS 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 the UHS inoperable due to inoperable fans and/or associated spray cells, restore to OPERABLE status within 72 hours; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the UHS inoperable due to an inoperable NSCW transfer pump, restore the transfer pump to OPERABLE status within 8 days or implement an alternate method of transfer of basin content; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Implementation of the alternate method of transfer of basin content shall not exceed 31 days. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE:

a. At least once per 24 hours by verifying the water level and water temperature to be within their limits.

b. At least once per 31 days by verifying that the required number of fans start and operate for at least 15 minutes.

c. The NSCW transfer pumps will be tested pursuant to the requirement of Specification 4.0.5.

TO BE REVISED TO REFLECT UNIT 2

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent Control Room Emergency Filtration Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 during movement of irradiated fuel or movement of loads over irradiated fuel.

ACTION:

MODES 1, 2, 3 or 4:

With one Control Room Emergency Filtration System inoperable, restore the inoperable system 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 during movement of irradiated fuel or movement of loads over irradiated fuel:

a. With one Control Room Emergency Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Emergency Filtration System in the emergency mode.

b. With both Control Room Emergency Filtration Systems inoperable, or with the OPERABLE Control Room Emergency Filtration System, required to be in the emergency mode by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving movement of irradiated fuel or movement of loads over irradiated fuel.

SURVEILLANCE REQUIREMENTS

4.7.6 Each Control Room Emergency Filtration System 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 80°F

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow (FI-12191, FI-12192) through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heater control circuit energized.

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 filtration system satisfies the in-place testing acceptance criteria of greater than or equal to 99.95% filter retention while operating the system at a flow rate of 19,000 cfm ±10% and performing the following tests:

(a) A visual inspection of the control room emergency filtration system shall be made before each DOP test or activated carbon adsorber section leak test in accordance with Section 5 of ANSI N510-1980.

(b) An in-place DOP test for the HEPA filters shall be performed in accordance with Section 10 of ANSI N510-1980.

(c) A charcoal adsorber section leak test with a gaseous halogenated hydrocarbon refrigerant shall be performed in accordance with Section 12 of ANSI N510-1980.

2) Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Section 13 of ANSI N510-1980 meets the laboratory testing criterion of greater than or equal to 99.8% when tested with methyl iodide at 30°C and 70% relative humidity.

3) Verifying a system flow rate of 19,000 cfm ± 10% during system operation when tested in accordance with Section 8 of ANSI N510-1980.

d. 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 Section 13 of ANSI N510-1980 meets the laboratory testing criterion of greater than or equal to 99.8% when tested with methyl iodide at 30°C and 70% relative humidity.

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters, charcoal adsorber banks and cooling coil is less than 7.1 inches Water Gauge while operating the system at a flow rate of 19,000 cfm ± 10%;

2) Verifying that on a Control Room Isolation Test Signal, the system automatically switches into an emergency mode of operation with flow through the HEPA filters and charcoal adsorber banks;

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge at less than or equal to a pressurization flow of 850 cfm relative to adjacent areas during system operation;

4) Verifying that the heaters dissipate 118 ± 6 kW when tested in accordance with Section 14 of ANSI N510-1980; and

5) Verifying that on a Control Room/Toxic Gas Isolation test signal, the control room isolation dampers close within 6 seconds and the system automatically switches into an isolation mode of operation with flow through the HEPA filters and charcoal adsorbers.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in place in accordance with Section 10 of ANSI N510-1980 while operating the system at a flow rate of 19,000 cfm ± 10%; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal absorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when tested in-place in accordance with Section 12 of ANSI N510-1980 while operating the system at a flow rate of 19,000 cfm ± 10%.

3/4.7.7 PIPING PENETRATION AREA FILTRATION AND EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent Piping Penetration Area Filtration and Exhaust Systems shall be OPERABLE.

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

ACTION:

With one Piping Penetration Area Filtration and Exhaust System inoperable, restore the inoperable system 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

4.7.7 Each Piping Penetration Area Filtration and Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow (FI-12629, FI-12542) through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heater control circuit energized;

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 that the cleanup system satisfies the in-place testing acceptance criteria of greater than or equal to 99.95% filter retention while operating the system at a flow rate of 15,500 cfm ± 10% (Unit 1), and ____ cfm ± 10% (Unit 2) performing the following tests:

(a) A visual inspection of the piping penetration area filtration and exhaust system shall be made before each DOP test or activated carbon adsorber section leak test in accordance with Section 5 of ANSI N510-1980.

(b) An in-place DOP test for the HEPA filters shall be performed in accordance with Section 10 of ANSI N510-1980.

(c) A charcoal adsorber section leak test with a gaseous halogenated hydrocarbon refrigerant shall be performed in accordance with Section 12 of ANSI N510-1980.

* The provisions of this specification are not applicable to Unit 2 until its initial entry into MODE 2.

3/4.7.7 PIPING PENETRATION AREA FILTRATION AND EXHAUST SYSTEM

SURVEILLANCE REQUIREMENTS (continued)

2) Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Section 13 of ANSI N510-1980 meets the laboratory testing criterion of greater than or equal to 99.8% when tested with methyl iodide at 30°C and 70% relative humidity.

3) Verifying a system flow rate of 15,500 cfm ± 10% (Unit 1) and ____ cfm ± 10% (Unit 2) during system operation when tested in accordance with Section 8 of ANSI N510-1980.

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 Section 13 of ANSI N510-1980 meets the laboratory testing criteria of greater than or equal to 99.8% when tested with methyl iodide at 30°C and 70% relative humidity;

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 6 inches Water Gauge while operating the system at a flow rate of 15,500 cfm ± 10% (Unit 1) and ____ cfm ± 10% (Unit 2).

2) Verifying that the system starts on a Containment Ventilation Isolation test signal,

3) Verifying that the system maintains the Piping Penetration Filtration Exhaust Unit Room at a negative pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere (PDIC-2550, PDIC-2551), and

4) Verifying that the heaters dissipate 80 ± 4 kW when tested in accordance with Section 14 of ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with Section 10 of ANSI N510-1980 while operating the system at a flow rate of 15,500 cfm ± 10% (Unit 1) and ____ cfm ± 10% (Unit 2).

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Section 12 of ANSI N510-1980 while operating the system at a flow rate of 15,500 cfm ± 10% (Unit 1) and ____ cfm ± 10% (Unit 2).

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements 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 4.7.8g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to 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 shall be performed~~ at the first refueling outage. ~~Otherwise,~~ Subsequent visual inspections shall be performed in accordance with the following schedule:

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%

*The inspection interval for 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.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) 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 that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.8f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

d. Transient Event Inspection

An 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 and a visual inspection of the systems 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.

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 for each type 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 4.7.8f., 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

e. Functional Tests (Continued)

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 4.7.8f. 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.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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 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.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up 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 4.7.8e. for snubbers not meeting the functional test acceptance criteria.

SURVEILLANCE REQUIREMENTS (Continued)

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 6.9.3.

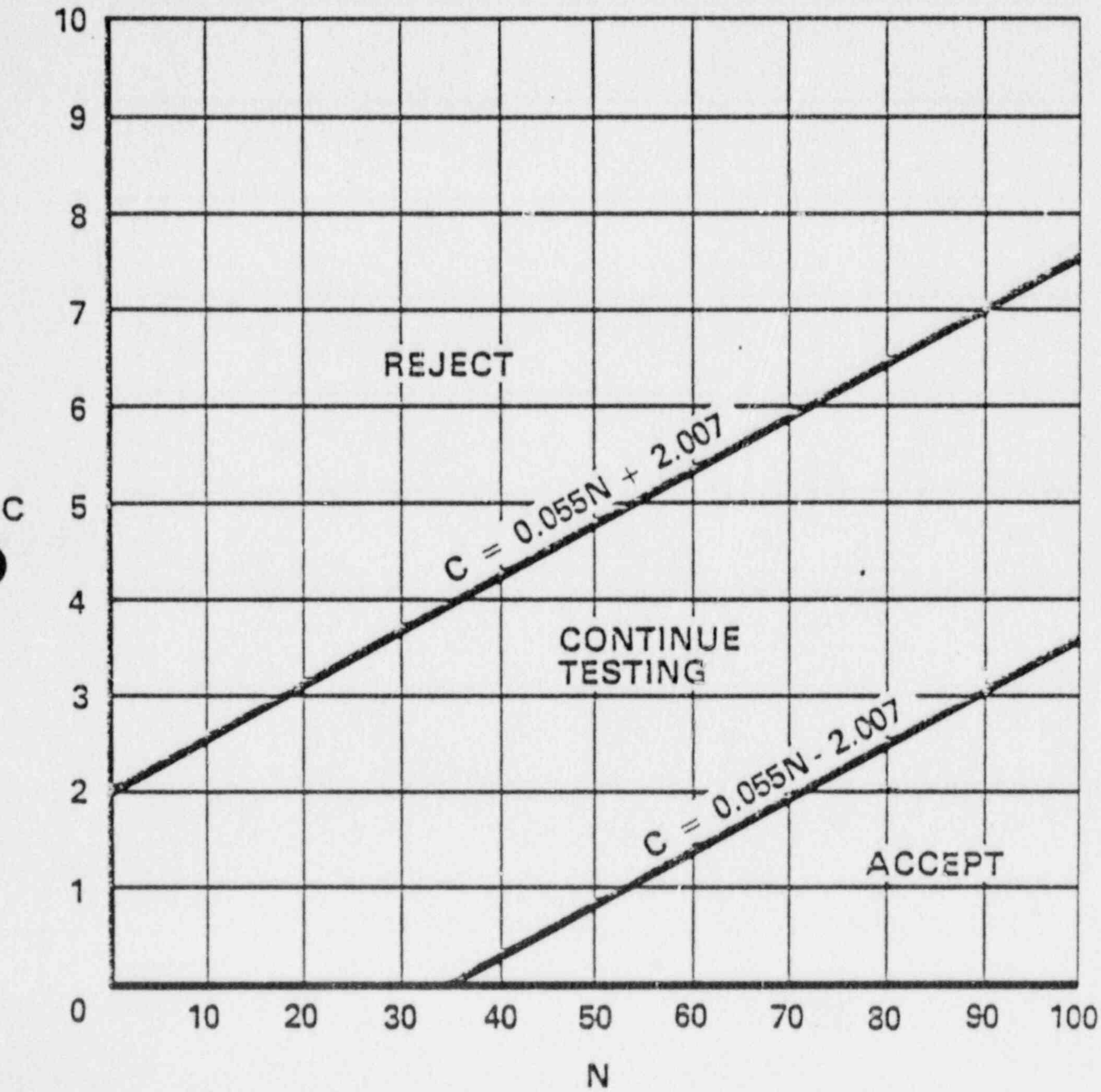

FIGURE 4.7-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 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

4.7.9.1 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.

4.7.9.2 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 the core and following repair or maintenance to the source.

3/4.7.10 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.10 The Maximum Normal Temperature of each area shown in Table 3.7-3 shall not be exceeded for more than 8 hours and the Maximum Abnormal Temperature of Table 3.7-3 shall not be exceeded.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the Maximum Normal Temperature limit(s) shown in Table 3.7-3 for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With one or more areas exceeding the Maximum Abnormal Temperature limit(s) shown in Table 3.7-3 prepare and submit a Special Report as required by ACTION a. above and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.10 The temperature in each of the areas shown in Table 3.7-3 shall be determined to be within its limit at least once per 12 hours.

TABLE 3.7-3

AREA TEMPERATURE MONITORING

BUILDING	ROOM NO	MAXIMUM NORMAL TEMP	MAXIMUM ABNORMAL TEMP
FUEL	B008	104	120
AUXILIARY	110	100	104
AUXILIARY	202.	100	107
AUXILIARY	203	100	107
AUXILIARY	A017	100	112
AUXILIARY	A047	100	107
AUXILIARY	B017	100	100
AUXILIARY	C113	100	109
AUXILIARY	C120	100	106
AUXILIARY	D053	100	110
AUXILIARY	D068	100	106
AUXILIARY	D072	100	105
AUXILIARY	D075	100	106
AUXILIARY	D119	100	105
AUXILIARY	D121	100	103
CONTROL	147	80	87
CONTROL	149	80	87
CONTROL	A054	100	114
CONTROL	A062	100	103
CONTROL	B065	100	106
CONTROL	B074	100	107
CONTROL	B078	100	104

Rooms associated with Unit 2 will be added later.

3/4 7.11 ENGINEERED SAFETY FEATURES (ESF) ROOM COOLER AND SAFETY-RELATED CHILLER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11 Two independent ESF room cooler and safety-related chiller trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one ESF room cooler and safety-related chiller train OPERABLE, restore the inoperable train 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

4.7.11 Two ESF room cooler and safety-related chiller trains 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.

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.

c. At least once per 18 months by verifying that each ESF room cooler fan and each train of safety-related chillers (pump and chiller) start automatically on a Safety Injection test signal.

3/4.7.12 REACTOR COOLANT PUMP THERMAL BARRIER COOLING WATER ISOLATION

LIMITING CONDITION FOR OPERATION

3.7.12 The reactor coolant pump thermal barrier isolation function shall be OPERABLE.

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

ACTION:

With the reactor coolant pump thermal barrier isolation function inoperable, restore 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

4.7.12 The reactor coolant pump thermal barrier isolation function shall be demonstrated OPERABLE by:

a. Verifying that valve HV-2041 automatically closes on thermal barrier outlet high header pressure and high header flow test signals at least once per 36 months on a STAGGERED TEST BASIS.

b. Verifying that valves HV-19051, 19053, 19055, and 19057 automatically close on thermal barrier outlet high flow test signals at least once per 18 months.

3/4.7.13 DIESEL GENERATOR BUILDING AND AUXILIARY FEEDWATER PUMPHOUSE ESF HVAC SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.13 The diesel generator building and auxiliary feedwater pumphouse ESF HVAC systems shall be OPERABLE with:

a. At least two ESF supply fans and associated dampers per diesel generator train, and

b. At least three ESF auxiliary feedwater pumphouse HVAC systems.

APPLICABILITY: Whenever the associated diesel generator or auxiliary feedwater pumps are required to be operable.

ACTION:

a. With 1 or more supply fans to a given diesel generator train inoperable, restore the inoperable fan(s) 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 1 or more ESF auxiliary feedwater pumphouse HVAC system inoperable, follow the ACTION specified in specification 3.7.1.2 for an inoperable auxiliary feedwater pump.

SURVEILLANCE REQUIREMENTS

4.7.13 The diesel generator building and auxiliary feedwater pumphouse ESF HVAC system shall be demonstrated OPERABLE:

a. At least once per 18 months by:

1) Verifying that the diesel generator ESF supply fans, ~~1~~ 1/2-1566-B7-001 (train A) and ~~1~~ 1/2-1566-B7-002 (train B) start automatically and the associated intake and discharge dampers actuate to their correct position on their train associated diesel generator running signal.

2) Verifying that diesel generator ESF supply fans ~~1~~ 1/2-1566-B7-003 and ~~1~~ 1/2-1566-B7-004 start automatically and the associated intake and discharge dampers actuate to their correct position on a high diesel generator building temperature signal coincident with diesel generator running signal.

b. At least once per 18 months by:

1) Verifying that the auxiliary feedwater pumphouse ESF supply fans, ~~1~~ 1/2-1593-B7-001 and ~~1~~ 1/2-1593-B7-002 and associated shutoff dampers actuate to their correct position on a high room temperature signal.

SURVEILLANCE REQUIREMENTS

4.7.13 (Continued)

2) Verifying that the ESF outside air intake and exhaust dampers for the turbine driven auxiliary feedwater pump actuate to the correct position on a turbine driven auxiliary feedwater pump automatic start signal.

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System, and

b. Two separate and independent diesel generators, each with:

1) A day tank containing a minimum volume of 650 gallons (52% of instrument span) of fuel (LI-9018, LI-9019),

2) A separate Fuel Storage System containing a minimum volume of 68,000 gallons of fuel (76% of instrument span) (LI-9024, LI-9025), and

3) A separate fuel transfer pump,

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

ACTION:

a. With one offsite circuit of the above-required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either diesel generator has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 for each such diesel generator, separately, within 24 hours unless the diesel generator is already operating. Restore the offsite circuit 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 either diesel generator inoperable, demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 24 hours*#. Restore the inoperable diesel generator 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.

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*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

#The diesel shall not be rendered inoperable by activities performed to support testing pursuant to the Action Statement (e.g., an air roll).

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. 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. offsite source by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter, and, if the diesel generator became inoperable due to any cause other than preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 8 hours*, unless the OPERABLE diesel generator is already operating#. 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 the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of 3.8.1.1, Action Statement a or b, as appropriate, with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source. A successful test of diesel generator OPERABILITY per Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 performed under the Action Statement for an OPERABLE diesel generator or a restored to OPERABLE diesel generator satisfies the diesel generator test requirement of Action Statement a or b.

d. With one diesel generator inoperable in addition to ACTION b. or c. above, verify that:

   1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

   2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater pump is 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.

e. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators separately by performing the requirements of Specification 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 8 hours#, 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. Following restoration of one offsite source, follow Action Statement a with the time requirement of that Action Statement based

*This test is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

#The diesel shall not be rendered inoperable by activities performed to support testing pursuant to the Action Statement (e.g., an air roll).

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

on the time of the initial loss of the remaining inoperable offsite a.c. circuit. A successful test(s) of diesel OPERABILITY per Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 performed under this Action Statement for the OPERABLE diesels satisfies the diesel generator test requirement for Action Statement a.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing the requirements of Specification 4.8.1.1.1.a. within 1 hour and at least once per 8 hours thereafter; restore at least one 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. Following restoration of one diesel generator unit, follow Action Statement b with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable diesel generator. A successful test of diesel OPERABILITY per Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 performed under this Action Statement for a restored to OPERABLE diesel satisfies the diesel generator test requirements of Action Statement b.

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, and indicated power availability.

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 day tank (LI-9018, LI-9019),

2) Verifying the fuel level in the fuel storage tank (LI-9024, LI-9025),

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank,

4) Verifying the diesel starts and that the generator voltage and frequency are 4160 + 170, -410 volts and 60 + 1.2 Hz within 11.4 seconds* after the start signal. The diesel generator shall be started for this test by using one of the following signals:

*All diesel generator starts for the purpose of surveillance testing as required by Specification 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

a) Manual, or

b) Simulated loss-of-offsite power by itself, or

c) Simulated loss-of-offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

5) Verifying the generator is synchronized, loaded to an indicated 6800-7000 kW*#, and operates at this loaded condition for at least 60 minutes, and

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

7) Verifying the pressure in at least one diesel generator airstart receiver (PI-9060, PI-9061, PI-9064, PI-9065) to be greater than or equal to 210 psig.

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 fuel tank;

c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

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:

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*This band is meant as guidance to avoid routine overloading of the diesel generator. Loads in excess of the band or momentary variations due to changing bus loads shall not invalidate the test.

#All diesel generator starts for the purpose of surveillance testing as required by Specification 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

SURVEILLANCE REQUIREMENTS (Continued)

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 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.

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;

f. At least once per 92 days and from new fuel prior to addition to the storage tank obtain a sample and verify that the neutralization number is less than 0.2 and the mercaptan content is less than 0.01% #,##.

g. At least once per 184 days by:

1) Verifying the diesel starts* from ambient conditions and the generator voltage and frequency are 4160 + 170, -410 volts and 60 ± 1.2 Hz within 11.4 seconds after the start signal. The diesel generator shall be started for this test by using one of the signals listed in Surveillance Requirement 4.8.1.1.2.a.4. This test, if it is performed so it concides with the testing required by Surveillance Requirement 4.8.1.1.2.a.4, may also serve to concurrently meet those requirements as well.

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*All engine starts for the purpose of surveillance testing as required by 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer to minimize mechanical stress on the diesel engine.

#Mercaptan content shall be required to be verified within specification for new fuel prior to its addition, for up to 15,000 gallons of fuel added to the tank, if the last tank sample had a mercaptan content of less than 0.007%. All subsequent new fuel addition will require mercaptan content verification prior to its addition until the tank contents are verified to be less than .007%.

##Neutralization number will not have to be verified less than 0.2, for new fuel prior to its addition, until 60 days after license issuance. Until that time verification of new fuel specifications will be completed within 30 days of addition.

2) Verifying the generator is synchonized, loaded to an indicated value of 6100 - 7000 kW*** in less than or equal to 60 seconds, and operates with a load of 6800-7000 kW*** for at least 60 minutes. This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.5, may also serve to concurrently meet those requirements as well.

h. 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 manufacturers' recommendations for this class of standby service;

2) Verifying the diesel generator capability to reject a load of greater than or equal to 671 kW (motor-driven auxiliary feedwater pump) while maintaining voltage at 4160 + 240, -410 volts and speed of less than 484 rpm (less than nominal speed plus 75% of the difference between nominal speed and the Overspeed Trip Setpoint); and recovering voltage to within 4160 + 170, -410 volts within 3 seconds.

3) Verifying the diesel generator capability to reject a load of 7000 kW without tripping. The generator voltage shall not exceed 5000 volts during and following the load rejection;

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization 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 11.5 seconds,* energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 +170, -410 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on an ESF Actuation 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 +170, -410 volts and 60 ± 1.2 Hz within 11.4 seconds after the

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*All engine starts for the purpose of surveillance testing as required by Specification 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer to minimize mechanical stress and wear on the diesel engine.

**For any start of a diesel, the diesel must be operated with a load in accordance with the manufacturer's recommendations.

***This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band or momentary variations due to changing bus loads shall not invalidate this test.

auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF Actuation test signal, and:

   a) Verifying deenergization 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 11.5 seconds,* energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 +170, -410 volts and 60 $\pm$ 1.2 Hz during this test; and

   c) Verifying that all automatic diesel generator trips, except engine overspeed, low lube oil pressure, high jacket water temperatures and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) 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 an indicated 7600 to 7700 kW,** and during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 6800-7000 kW.** The generator voltage and frequency shall be 4160 + 170, - 410 volts and 60 ± 1.2 Hz within 11.4 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 4.8.1.1.2h.6)b);##

8) Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 7000 kW;

9) Verifying the diesel generator's capability to:

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*All engines starts for the purpose of surveillance testing as required by 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer to minimize mechanical stress and wear on the diesel engine.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band or momentary variations due to changing bus loads shall not invalidate the test.

#Failure to maintain voltage and frequency requirements due to grid disturbances does not render a 24-hour test as a failure.

##If Specification 4.8.1.1.2h.6)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at the load required by Surveillance Requirement 4.8.1.1.2.a5 kW for 1 hour or until operating temper ature has stabilized.

a) Synchronize 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.

10) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines;

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within $\pm$ 10% of its design interval;

i. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 440 rpm in less than or equal to 11.4 seconds; and

j. 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 solution, or equivalent, and

2) Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported to the Commission in a Special Report pursuant to Specification 6.8.2 within 30 days. 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 in the last 100 valid tests on a per nuclear unit basis 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.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 20 Valid Tests*	Number of Failures in Last 100 Valid Tests*	Test Frequency
$\leq$ 1	$\leq$ 4	Once per 31 days
$\geq$ 2**	$\geq$ 5	Once per 7 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, but determined on a per diesel generator basis.

For the purposes of determining the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul to like-new condition is completed, provided that the overhaul, including appropriate post-maintenance operation and testing, is specifically approved by the manufacturer and if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with the routine Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 and four tests in accordance with the 184-day testing requirement of Surveillance Requirement 4.8.1.1.2.f. If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

**The associated test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to one.

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) A day tank containing a minimum volume of 650 gallons (52% of instrument span) (LI-9018, LI-9019) of fuel,

2) A fuel storage system containing a minimum volume of 68,000 gallons of fuel (76% of instrument span) (LI-9024, LI-9025), and

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 crane operation with loads over the fuel storage pool, and provide relief capability for the Reactor Coolant System in accordance with Specification 3.4.9.3. 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 (except for Specification 4.8.1.1.2a.5), and 4.8.1.1.3.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125 V-dc Battery bank ~~1~~1/2AD1B, and one of its associated full-capacity chargers.

b. 125 V-dc Battery bank ~~1~~1/2BD1B, and one of its associated full-capacity chargers.

c. 125 V-dc Battery bank ~~1~~1/2CD1B, and one of its associated full-capacity chargers.

d. 125 V-dc Battery bank ~~1~~1/2DD1B, and one of its associated full-capacity chargers.

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

ACTION:

With less than the above minimum required D.C. electrical sources OPERABLE restore the inoperable D.C. electrical source(s) 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

4.8.2.1 Each 125-volt battery bank and one associated charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 126 volts on float charge.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 109.7 volts for trains A&B, 108.3 volts for train C, and 106.2 volts for train D at a battery room minimum temperature of 70°F, or battery overcharge with battery terminal voltage above 140 volts, by verifying that:

   1) The parameters in Table 4.8-2 meet the Category B limits,

   2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 50×10^{-6} ohm, and

   3) The average electrolyte temperature of twelve connected cells is above 70° 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 anticorrosion material,

   3) The resistance of each cell-to-cell and terminal connection is less than or equal to 50×10^{-6} ohm, and

   4) The battery charger will supply at least 400 amperes for system A and B, 300 amperes for system C, and 200 amperes for system D at 125 volts nominally for at least 8 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 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. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity 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.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
PARAMETER	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)	> 2.10 volts
Specific Gravity(4)	≥ 1.195(5)	≥ 1.190	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > 1.200	Average of all connected cells ≥ 1.190(5)

TABLE NOTATIONS

(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.

(4) Corrected for electrolyte temperature (reference temperature of 77°F) and level.

(5) Or battery charging current is less than 2 amps when on float charge.

(6) Corrected for average electrolyte temperature.

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one train related pair of 125-V battery banks (either 125-V battery banks 1AD1B and 1CD1B or 125-V battery banks 1BD1B and 1DD1B) and one associated full-capacity charger per required battery bank shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With the required battery bank and/or both chargers 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 at least one charger to OPERABLE status as soon as possible, and within 8 hours, provide relief capability for the Reactor Coolant System in accordance with Specification 3.4.9.3.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery bank and full-capacity charger shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open between redundant busses within the unit:

a. A.C. Emergency Busses consisting of:
   1. Train A
      a) 4160 volt switchgear ~~1~~AA02 1/2
      b) 480 volt switchgear ~~1~~AB04 1/2
         1) MCC ~~1~~ABE 1/2
      c) 480 volt switchgear ~~1~~AB05 1/2
         1) MCC ~~1~~ABA 1/2
         2) MCC ~~1~~ABC 1/2
         3) MCC ~~1~~ABF 1/2
      d) 480 volt switchgear ~~1~~AB15 1/2
         1) MCC ~~1~~ABB 1/2
         2) MCC ~~1~~ABD 1/2
   2. Train B
      a) 4160 volt switchgear ~~1~~BA03 1/2
      b) 480 volt switchgear ~~1~~BB06 1/2
         1) MCC ~~1~~BBE 1/2
      c) 480 volt switchgear ~~1~~BB07 1/2
         1) MCC ~~1~~BBA 1/2
         2) MCC ~~1~~BBC 1/2
         3) MCC ~~1~~BBF 1/2
      d) 480 volt switchgear ~~1~~BB16 1/2
         1) MCC ~~1~~BBB 1/2
         2) MCC ~~1~~BBD 1/2

b. 120 volt A.C. vital Busses
   1. Associated with Train A
      a) Channel I
         1) Panel ~~1~~AY1A 1/2 energized from inverter ~~1~~AD1I1 1/2 connected to switchgear ~~1~~AD1* 1/2
         2) Panel ~~1~~AY2A 1/2 energized from inverter ~~1~~AD1I11 1/2 connected to switchgear ~~1~~AD1* 1/2
      b) Channel III
         1) Panel ~~1~~CY1A 1/2 energized from inverter ~~1~~CD1I3 1/2 connected to switchgear ~~1~~CD1* 1/2
   2. Associated with Train B
      a) Channel II
         1) Panel ~~1~~BY1B 1/2 energized from inverter ~~1~~BD1I2 1/2 connected to switchgear ~~1~~BD1* 1/2

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*Two inverters in a single train may be disconnected from their associated switchgear for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided: (1) their associated panels are energized from their regulated transformers, and (2) the panels associated with the other battery bank powered from that AC train are energized in the specified manner.

LIMITING CONDITION FOR OPERATION

2) Panel 1BY2B energized from inverter 1BD1I12 connected to switchgear 1BD1*

b) Channel IV

1) Panel 1DY1B energized from inverter 1DD1I4 connected to switchgear 1DD1*

c. 125 volt D.C. Busses consisting of:

1. Associated with Train A

a) System A

1) Switchgear 1AD1 energized from battery 1AD1B
2) MCC 1AD1M energized from switchgear 1AD1
3) Distribution panel 1AD11 energized from switchgear 1AD1
4) Distribution panel 1AD12 energized from switchgear 1AD1

b) System C

1) Switchgear 1CD1 energized from battery 1CD1B
2) MCC 1CD1M energized from switchgear 1CD1
3) Distribution panel 1CD11 energized from switchgear 1CD1

2. Associated with Train B

a) System B

1) Switchgear 1BD1 energized from battery 1BD1B
2) MCC 1BD1M energized from switchgear 1BD1
3) Distribution panel 1BD11 energized from switchgear 1BD1
4) Distribution panel 1BD12 energized from switchgear 1BD1

b) System D

1) Switchgear 1DD1 energized from battery 1DD1B
2) Distribution panel 1DD11 energized from switchgear 1DD1

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

ACTION:

a. With one of the required trains of A.C. emergency busses not fully energized, reenergize the train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Two inverters in a single train may be disconnected from their associated switchgear for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided: (1) their associated panels are energized from their regulated transformers, and (2) the panels associated with the other battery bank powered from that AC train are energized in the specified manner.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. With one 120 volt A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated switchgear: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the panel 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 125 volt D.C. bus ,MCC, not energized in the specified manner, reenergize the switchgear or distribution panel in the specified manner 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 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.

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses* shall be energized in the specified manner:

a. One train of A.C. emergency busses consisting of one 4160-volt switchgear, three 480-volt A.C. switchgear, and six 480-volt A.C. Motor Control Centers.

b. One train of 120-volt A.C. vital busses energized from their associated inverters connected to their respective D.C. busses, and

c. One train of 125-volt D.C. switchgear and associated distribution equipment energized from its associated battery bank.

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, and within 8 hours, provide relief capability for the ~~RCS~~ REACTOR COOLANT SYSTEM in accordance with Specification 3.4.9.3.

SURVEILLANCE REQUIREMENTS

4.8.3.2 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.

*All electrical busses shall be from the same train.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES AND FEEDER BREAKERS TO ISOLATION TRANSFORMERS BETWEEN 480 V CLASS 1E BUSSES AND NON CLASS 1E EQUIPMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices and feeder breakers to isolation transformers between 480 V Class 1E busses and non Class 1E equipment shall be OPERABLE.

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

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) or feeder breakers to isolation transformers between 480 V Class 1E busses and non Class 1E equipment inoperable:

a. Restore the protective device or feeder breaker to OPERABLE status or deenergize the circuit(s) by racking out, locking open, or removing the inoperable circuit breaker or protective device and tripping the associated backup circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the inoperable circuit breaker or protective device racked out, locked open or removed at least once per 31 days threafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices or feeder breakers in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

b. Deenergize the circuits by racking out, locking open, or removing the inoperable circuit breaker or protective device within 72 hours, declare the affected system or component inoperable, and verify the inoperable circuit breaker or protective device racked out, locked open or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices or feeder breakers in circuits which have their inoperable circuit breaker racked out, or removed, or

c. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices and feeder breakers to isolation transformers between 480 V Class 1E busses and non Class 1E equipment shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1) By verifying that the 13.8 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least one of the circuit breakers, and performing the following:

SURVEILLANCE REQUIREMENTS (Continued)

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 breakers and control circuits function as designed, and

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least one of 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 these circuit breakers shall consist of injecting a current for each over current trip on the breaker and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is within the tolerances specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. 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

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.

SAFETY-RELATED MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection bypass devices of each safety-related motor-operated valve given in Table 3.8-1 shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection bypass devices for any one or more safety-related motor-operated valves inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) of the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The above required thermal overload protection bypass devices shall be verified to be OPERABLE.

a. Following maintenance on the valve motor starter, and

b. Following any periodic testing during which the thermal overload device was temporarily placed in force.

c. At least once per 18 months, during shutdown.

TABLE 3.8-1

SAFETY-RELATED
MOTOR OPERATED VALVES THERMAL OVERLOAD
PROTECTION BYPASS DEVICES

VALVE NUMBER	FUNCTION
1/2 HV-1668A	Nuclear Service Clg. Twr. A Return
1/2 HV-1668B	Nuclear Service Clg. Twr. A Fans Bypass
1/2 HV-1669A	Nuclear Service Clg. Twr. B Return
1/2 HV-1669B	Nuclear Service Clg. Twr. B Fans Bypass
1/2 HV-1806	CTMT Air Cool A7001/A7002 CW Inlet
1/2 HV-1807	CTMT Air Cool A7003/A7004 CW Inlet
1/2 HV-1808	CTMT Air Cool A7005/A7006 CW Inlet
1/2 HV-1809	CTMT Air Cool A7007/A7008 CW Inlet
1/2 HV-1822	CTMT Air Cool A7001/A7002 CW Outlet
1/2 HV-1823	CTMT Air Cool A7003/A7004 CW Outlet
1/2 HV-1830	CTMT Air Clr A7005/A7006 CW Outlet
1/2 HV-1831	CTMT Air Clr A7007/A7008 CW Outlet
1/2 HV-1974	Aux Comp CW Trn B Return Iso
1/2 HV-1975	Aux Comp CW Trn A Return Iso
1/2 HV-1978	Aux Comp CW Trn B Supply Iso
1/2 HV-1979	Aux Comp CW Trn A Supply Iso
1/2 HV-2041	Reactor Coolant Pumps Thermal Barrier ACCWS Outlet Header
1/2 HV-2134	Reactor Cavity Clg Coil E7001 Inlet Iso
1/2 HV-2135	Reactor Cavity Clg Coil E7002 Inlet Iso
1/2 HV-2138	Reactor Cavity Clg Coil E7001 Outlet Iso
1/2 HV-2139	Reactor Cavity Clg Coil E7002 Outlet Iso
1/2 HV-12114	CR Outside Air Intake Iso
1/2 HV-12115	CR Outside Air Intake Iso
1/2 HV-12118	CB CR Filter Units N7001 Inlets
1/2 HV-12119	CB CR Filter Units N7002 Inlets
1/2 HV-12128	CB CR Filter Units N7001 Outlets
1/2 HV-12129	CB CR Filter Units N7002 Outlets
1/2 HV-12130	CB CR Return Air Fans B7005 Inlets
1/2 HV-12131	CB CR Return Air Fans B7006 Inlets
1/2 HV-12727	CB SR Battery Rm Exh B7002 Damper
1/2 HV-12742	CB SF Battery Rm Exh B7001 Damper
1/2 HV-12748	CB SF Battery Rm Exh B7003 Damper
1/2 HV-12749	CB SF Battery Rm Exh B7004 Damper
1/2 HV-19051	Thermal Barrier Cooling Wtr RCP 001
1/2 HV-19053	Thermal Barrier Cooling Wtr RCP 002
1/2 HV-19055	Thermal Barrier Cooling Wtr RCP 003
1/2 HV-19057	Thermal Barrier Cooling Wtr RCP 004
1/2 HV-8920	Safety-Injection Pump Miniflow Isolation
1/2 HV-2624A	CTB Post LOCA Purge Exhaust Iso
1/2 HV-2624B	CTB Post LOCA Purge Exhaust Iso
1/2 HV-2626A	CTMT Bldg Norm Purge Supply Iso

TABLE 3.8-1

SAFETY-RELATED
MOTOR OPERATED VALVES THERMAL OVERLOAD
PROTECTION BYPASS DEVICES

VALVE NUMBER	FUNCTION
1/2 ~~1~~HV-2627A	CTB Norm Purge Supply Iso
1/2 ~~1~~HV-2628A	CTMT Bldg Norm Purge Exhaust Iso
1/2 ~~1~~HV-2629A	CTB Norm Purge Exhaust Iso
1/2 ~~1~~HV-5106	Aux FDW Pump Turbine
1/2 ~~1~~HV-5113	Conds Stor TK V4002 to Pump P4001
1/2 ~~1~~HV-5118	Conds Stor TK V4002 to Pump P4002
1/2 ~~1~~HV-5119	Conds Stor TK V4002 to Pump P4003
1/2 ~~1~~HV-5120	Aux FDW Pump P4001 Discharge Trn C
1/2 ~~1~~HV-5122	Aux FDW Pump P4001 Discharge Trn C
1/2 ~~1~~HV-5125	Aux FDW Pump P4001 Discharge Trn C
1/2 ~~1~~HV-5127	Aux FDW Pump P4001 Discharge Trn C
1/2 ~~1~~HV-5132	Aux FDW Pump P4002 Discharge Trn B
1/2 ~~1~~HV-5134	Aux FDW Pump P4002 Discharge Trn B
1/2 ~~1~~HV-5137	Aux FDW Pump P4003 Discharge Trn A
1/2 ~~1~~HV-5139	Aux FDW Pump P4003 Discharge Trn A
1/2 ~~1~~FV-5154	Aux FDW Pump P4002 Miniflow
1/2 ~~1~~FV-5155	Aux FDW Pump P4003 Miniflow
1/2 ~~1~~HV-8438	Charging Pump B Discharge
1/2 ~~1~~HV-8471A	Alt Charging Pump A Suction
1/2 ~~1~~HV-8471B	Alt Charging Pump B Suction
1/2 ~~1~~HV-8485A	Charging Pump A Discharge
1/2 ~~1~~HV-8485B	Charging Pump B Discharge
1/2 ~~1~~HV-8508A, B	Charging Pump Miniflow Iso to RWST
1/2 ~~1~~HV-8509A, B	Charging Pump Miniflow Iso to RWST
1/2 ~~1~~HV-9380A	CTMT ATM Unit 1 SVCE Air
1/2 ~~1~~HV-9380B	CTMT ATM Unit 1 SVCE Air
1/2 ~~1~~HV-12005	Trn B Aux FDW Pump Rm Inlet Damper
1/2 ~~1~~HV-12006	Trn A Aux FDW Pump Rm Inlet Damper
1/2 ~~1~~HV-12050	DGB Exh Fan B7001 Disch Damper (Trn A)
1/2 ~~1~~HV-12051	DGB Exh Fan B7003 Disch Damper (Trn A)
1/2 ~~1~~HV-12052	DGB Exh Fan B7005 Disch Damper (Trn A)
1/2 ~~1~~HV-12053	DGB Exh Fan B7002 Disch Damper (Trn B)
1/2 ~~1~~HV-12054	DGB Exh Fan B7004 Disch Damper (Trn B)
1/2 ~~1~~HV-12055	DGB Exh Fan B7006 Disch Damper (Trn B)
1/2 HV-8000A, B	PORV Blockline
1/2 HV-8147	Reg. Hx Tube Outlet to RCS Alternate Chg
1/2 HV-8146	Reg. Hx Tube Outlet to RCS Normal Chg
1/2 HV-8100, 8112	No 1 Seal Leakoff
1/2 HV-8103A, B, C, D	RCP No 1 Seal from Chg
1/2 HV-8111A, B, 8110	Chg Pump Miniflow
1/2 LV-0112C, B	VCT Discharge Header
1/2 LV-0112E, D	SIS RWST Discharge to Chg/SI Pump Suction
1/2 HV-8104	CVCS Boric Acid Filler to Charging Pump Suction

TABLE 3.8-1

SAFETY-RELATED
MOTOR OPERATED VALVES THERMAL OVERLOAD
PROTECTION BYPASS DEVICES

	VALVE NUMBER	FUNCTION
1/2	HV-8105; 8106	Chg Pump to RCS Isolation
1/2	HV-8807A, C; 8924	HHSI Suction to Chg/SI Suction
1/2	HV-8801A, B	BIT Discharge
1/2	HV-8808A, B, C, D	Accumulator Discharge
1/2	HV-8811A, B	Containment Emergency Sump Isolation
1/2	HV-8812A, B	RHR Suction from RWST
1/2	HV-8809A, B	RHR Discharge Header
1/2	HV-8804A	RHR Hx No. 1 Outlet to Charge Pump
1/2	HV-8804B	RHR Hx No. 2 Outlet to SI Pumps
1/2	HV-8806	RWST Discharge Header to SI Pumps
1/2	HV-8923A, B	SI Pump Suction Isolation
1/2	HV-8813; 8814	SI Pump Miniflow
1/2	HV-8821A, B	SI Pump Crosschannel
1/2	HV-8835	SI Pump Discharge to Cold Legs
1/2	HV-8840	RHR Pump Discharge to Hot Legs
1/2	HV-8802A, B	SI Pump Discharge Header
1/2	HV-8701A, B; 8702A, B	RHR Suction from RCS Hot Legs 1, 4
1/2	FV-0610; 0611	RHR Miniflow
1/2	HV-8716A, B	RHR Cross Connect
1/2	HV-9002A, B	Spray Pump Containment Emergency Sump Isolation
1/2	HV-9003A, B	Spray Pump Containment Emergency Sump Isolation
1/2	HV-9017A, B	Spray Pump Suction from RWST
1/2	HV-9001A, B	Spray Pump Discharge Header
1/2	HV-8994A, B	Spray Additive Tank Discharge
1/2	HV-11600	NSCW Pump Discharge Isolation
1/2	HV-11605	NSCW Pump Discharge Isolation
1/2	HV-11606	NSCW Pump Discharge Isolation
1/2	HV-11607	NSCW Pump Discharge Isolation
1/2	HV-11612	NSCW Pump Discharge Isolation
1/2	HV-11613	NSCW Pump Discharge Isolation
1/2	PV-2550A	Piping Penetration Room to Atmosphere
1/2	PV-2551A	Piping Penetration Room to Atmosphere
1/2	HV-3009	TDAFP Steam Supply Isolation
1/2	HV-3019	TDAFP Steam Supply Isolation
1/2	HV-8116	Charging Pump Discharge Boron Injection
1/2	PV-15129	TDAFP Trip and Throttle Valve
1/2	HV-2582A	CTB Cooling Unit A7001
1/2	HV-2582B	CTB Cooling Unit A7002
1/2	HV-2583A	CTB Cooling Unit A7003
1/2	HV-2583B	CTB Cooling Unit A7004
1/2	HV-2584A	CTB Cooling Unit A7005
1/2	HV-2584B	CTB Cooling Unit A7006
1/2	HV-2585A	CTB Cooling Unit A7007
1/2	HV-2585B	CTB Cooling Unit A7008

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 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 are met:

a. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2000 ppm.*

Additionally, valves 1208-U4-175, 1208-U4-177, 1208-U4-183, and 1208-U4-176 shall be closed and secured in position.

APPLICABILITY: MODE 6.

ACTION:

a. With the requirements of a. and b. above not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 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.

b. With valves 1208-U4-175, 1208-U4-177, 1208-U4-183, and 1208-U4-176 not closed and secured in position, immediately close and secure in position.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.2 Valves 1208-U4-175, 1208-U4-177, 1208-U4-183, and 1208-U4-176 shall be verified closed and secured in position by mechanical stops at least once per 31 days.

* During initial fuel load for Unit 2, the boron concentration limitation for the Unit 2 refueling canal is not applicable provided the Unit 2 refueling canal level is verified to be below the reactor vessel flange elevation at least once per 12 hours.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors (NI-0031B,D&E, NI-0032B,D&G) shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and 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 (NI-0031B,D&E, NI-0032B,D&G) shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor 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 vessel.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building 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 (HV-2626 A&B, HV-2627 A&B, HV-2628 A&B, HV-2629 A&B).

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 building 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 (HV-2626 A&B, HV-2627 A&B, HV-2628 A&B, HV-2629 A&B) 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 building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment ventilation isolation valves per the applicable portions of Specification 4.6.3.2.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION 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 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of fuel assemblies and/or rod control cluster assemblies or thimble plug assemblies, and the auxiliary hoist shall be used for latching and unlatching, or handling of control rod drive shafts. The refueling machine and auxiliary hoist shall be OPERABLE with:

a. The refueling machine having:

   1) A design rated load of 3166 pounds, and

   2) An overload cutoff limit less than or equal to the design rated load.

b. The auxiliary hoist having:

   1) A minimum capacity of 3000 pounds, and

   2) A load indicator which shall be used to administratively prevent lifting loads in excess of 1000 pounds.

APPLICABILITY: During movement of fuel assemblies, rod control cluster assemblies, thimble plug assemblies, or control rod drive shafts within the reactor vessel.

ACTION:

With the requirements for the refueling machine or auxiliary hoist OPERABILITY not satisfied, suspend use of the inoperable refueling machine or auxiliary hoist from operations involving the movement of fuel assemblies, rod control cluster assemblies, thimble plug assemblies, or control rod drive shafts within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The refueling machine shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 125% of the design rated load and demonstrating an automatic load cutoff prior to the load exceeding the design rated load.

4.9.6.2 The auxiliary hoist and associated load indicator shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 1250 pounds.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2300 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With irradiated fuel assemblies in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

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

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2300 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

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 greater than or equal to 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 (FIC-0618A, FIC-0619A) 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.

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 greater than or equal to 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 (FIC-0618A, FIC-0619A) 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 2-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

LIMITING CONDITION 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.

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 each of the containment radiation monitoring instrumentation channels (RE-0002, RE-0003, RE-2565).

3/4.9.10 WATER LEVEL - REACTOR VESSEL

FUEL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies within the containment when the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies within the containment.

SURVEILLANCE REQUIREMENTS

4.9.10.1 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.

WATER LEVEL - REACTOR VESSEL

CONTROL RODS

LIMITING CONDITION FOR OPERATION

3.9.10.2 At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies within the reactor pressure vessel.

APPLICABILITY: During movement of control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirement of the above specification not satisfied, suspend all operations involving movement of control rods within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.2 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 control rods within the reactor vessel.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION 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 storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations over the spent fuel pool and restore the water level to within its limit within 4 hours.

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

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage 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 fuel storage pool.

3/4.9.12 FUEL HANDLING BUILDING POST ACCIDENT VENTILATION SYSTEM (Common System)

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Handling Building Post Accident Ventilation Systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in ~~the~~ either storage pool.

ACTION:

a. With one Fuel Handling Building Post Accident Ventilation System inoperable, fuel movement within ~~the~~ either storage pool or crane operation with loads over ~~the~~ either storage pool may proceed provided the OPERABLE Fuel Handling Building Post Accident 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 Post Accident Ventilation System OPERABLE, suspend all operations involving movement of fuel within ~~the~~ either storage pool or crane operation with loads over ~~the~~ either storage pool until at least one Fuel Handling Building Post Accident Ventilation System is restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Handling Building Post Accident Ventilation Systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heater control circuit energized;

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:

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the cleanup system satisfies the in-place testing acceptance criteria of greater than or equal to 99.0% filter retention while operating the system at a flow rate of 5000 cfm ± 10%, (FI-12551, FI-12552) and performing the following tests;

   (a) A visual inspection of the Fuel Handling Building Post Accident Ventilation System shall be made before each DOP test or activated carbon adsorber section leak test in accordance with Section 5 of ANSI N510-1980.

   (b) An in-place DOP test for the HEPA filters shall be performed in accordance with Section 10 of ANSI N510-1980.

   (c) A charcoal adsorber section leak test with a gaseous halogenated hydrocarbon refrigerant shall be performed in accordance with Section 12 of ANSI N510-1980.

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Section 13 of ANSI N510-1980, meets the laboratory testing criteria of greater than or equal to 99.8% when tested with methyl iodide at 30°C and 70% relative humidity; and

3) Verifying a system flow rate of 5000 cfm ± 10% during system operation when tested in accordance with Section 8 of ANSI N510-1980.

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 Section 13 of ANSI N510-1980 meets the laboratory testing criteria of greater than or equal to 99.8% when tested with methyl iodide at 30°C and 70% relative humidity.

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 6 inches Water Gauge while operating the system at a flow rate of 5000 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,

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that the system maintains the spent fuel storage pool area at a slightly negative pressure relative to t:e outside atmosphere during system operation, and

4) Verifying that the heaters dissipate 18 ± 2 kW when tested in accordance with Section 14 of ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when tested in-place in accordance with Section 10 of ANSI N510-1980 while operating the system at a flow rate of 5000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal absorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Section 12 of ANSI N510-1980 while operating the system at a flow rate of 5000 cfm ± 10%.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION 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 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 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all 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 30 gpm of a solution containing greater than or equal to 7000 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 control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each 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.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION 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 Specification 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 shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 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 CONDITION 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 channels are set in accordance with Table 2.2-1, Functional Units 5 and 2b, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°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 541°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 OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) (TI-0412, TI-0422, TI-0432, TI-0442) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specifications 3.4.1.2 may be suspended during the performance of natural circulation testing, or of hot rod drop time measurements in MODE 3 provided at least two reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

APPLICABILITY: During performance of natural circulation testing or hot rod drop time measurements.

ACTION:

With less than the above required reactor coolant loops OPERABLE during performance of natural circulation testing, or hot rod drop time measurements, immediately open the reactor trip breakers and comply with the provisions of the ACTION statements of Specification 3.4.1.2.

SURVEILLANCE REQUIREMENTS

4.10.4 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to initiation of natural circulation testing or hot rod drop time measurements and at least once per 4 hours during natural circulation testing or hot rod drop time measurements by verifying correct breaker alignments and indicated power availability and by verifying secondary side wide range water level to be greater than or equal to 17%.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual 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.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required 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 rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

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*This requirement is not applicable during the initial calibration of the Digital Rod Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-1) 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} microcurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

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

SURVEILLANCE REQUIREMENTS

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 ODCM 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)(1) (μCi/ml)
1. Batch Waste Release Tanks(2)	P Each Batch	P Each Batch	Principal Gamma Emitters(3)	5×10^{-7}
a. Waste-Monitor Tank 1901-T6-009			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
b. Waste-Monitor Tank 1901-T6-010				
	P Each Batch	M Composite(4)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
c. Drainage of Systems				
	P Each Batch	Q Composite(4)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
2. Continuous Releases(5)	Continuous(6)	W Composite(6)	Principal Gamma Emitters(3)	5×10^{-7}
			I-131	1×10^{-6}
a. Waste Water retention basin	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Continuous(6)	M Composite(6)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Continous(6)	Q Composite(6)	Sr-89, Sr-90	1×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.11-1 (Continued)

TABLE NOTATIONS

(1)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 system 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:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (sec).

Typical values of E, V, Y, and Δt should 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.

(2)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 ODCM to assure representative sampling.

TABLE 4.11-1 (Continued)

TABLE NOTATION (Continued)

(3) The principal gamma emmiters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. Ce-144 shall also be measured, but with an LLD of 5×10^{-6}. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.8.1.4 in the in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(4) 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 that is representative of the liquids released.

(5) A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release. This will not be considered a continuous release point until there is a confirmed primary to secondary release, it will be a continuous release until the leak is repaired. This surveillance will continue until three consecutive weekly composite samples show no activity above LLD.

(6) To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems 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, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases 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 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 ODCM at least once per 31 days.

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-1) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Radwaste Treatment System not in operation, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, a Special Report that includes the following information:

   1. 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 methodology and parameters in the ODCM when Liquid Radwaste Treatment Systems are not being fully utilized.

4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outside temporary tank 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 outside temporary tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, 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 6.8.1.4.

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 tanks shall be determined to be within the above limit by analyzing a representative sample of either the tank's contents at least once per 7 days when radioactive materials are being added to the tank or each batch of radioactive material prior to its addition to the tank.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

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

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 methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to 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 methodology and parameters in the ODCM 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)[1] (µCi/ml)
1. Waste Gas Decay Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[2]	1×10^{-4}
2. Containment Purge 24" or 14"	P Each PURGE[3] Grab Sample	P Each PURGE[3]	Principal Gamma Emitters[2]	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
3. a. Plant Vent	M[3], [4], [5] Grab Sample	M[3]	Principal Gamma Emitters[2]	1×10^{-4}
			H-3 (oxide)	1×10^{-6}
b. Condenser Air Ejector & Steam Packing Exhaust*	M[8] Grab Sample	M	Principal Gamma Emitters[2]	1×10^{-4}
			H-3 (oxide)	1×10^{-6}
4. All Release Types as listed in 3 above*	Continuous[6]	W[7] Charcoal Sample	I-131	1×10^{-12}
	Continuous[6]	W[7] Particulate Sample	Principal Gamma Emitters[2]	1×10^{-11}
	Continuous[6]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[6]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}

TABLE 4.11-2 (Continued)

TABLE NOTATIONS

* 3b may be obmitted provided the absence of a primary to secondary leak has been demonstrated; that is, the gamma activity in the secondary water does not exceed background by more than 20%.

(1) 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 system 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:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and time of counting (sec).

Typical values of E, V, Y, and Δt should 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 NOTATIONS (Continued)

(2)The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141 and Ce-144 in Iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.8.1.4 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(3)Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period. This requirement does not apply if (1) analysis shows that the Dose Equivalent I-131 concentration in the primary 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.

(4)Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

(5)Tritium grab samples shall be taken at least once per 7 days from the plant vent whenever spent fuel is in the spent fuel pool.

(6)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)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% of RATED THERMAL POWER within a 1-hour period and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs 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.

(8)Sampling not required if the gamma activity in the secondary water does not exceed background by more than 20%.

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases 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 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases 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

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The VENTILATION EXHAUST TREATMENT SYSTEM and the GASEOUS WASTE PROCESSING SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, a Special Report that includes the following information:

   1. 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

4.11.2.4 Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Gaseous Radwaste Treatment Systems are not being fully utilized.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the GASEOUS WASTE PROCESSING 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 WASTE PROCESSING SYSTEM greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the GASEOUS WASTE PROCESSING SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend oxygen addition and all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% 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 concentrations of hydrogen and oxygen in the GASEOUS WASTE PROCESSING SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the GASEOUS WASTE PROCESSING SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-10 of Specification 3.3.3.10.

RADIOACTIVE EFFLUENTS

GAS DECAY 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 2.0×10^5 curies of 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, 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 6.8.1.4.

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 have been added to the tank during the previous 24 hours.

3/4.11.3 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:

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 6.12, 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.

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

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 3.11.1.2a., 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 unit (including outside storage tanks etc.) 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 6.8.2, 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 20.405(c), 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.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 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 methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from ~~the~~ each unit (including outside storage tanks etc.) shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in ACTION a. of Specification 3.11.4.

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, prepare and submit to the Commission, in the Annual Radiological Environmental Surveillance Report required by Specification 6.8.1.3, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the confirmed** 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, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, 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:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

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 a MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specification 3.11.1.2, 3.11.2.2, 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 Surveillance Report required by Specification 6.8.1.3.

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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.

**A confirmatory reanalysis of the original, a duplicate, or a new sample may be desirable or appropriate. The results of the confirmatory analysis shall be completed at the earliest time consistent with the analysis but in any case within 30 days.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With milk or vegetation samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.13, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM 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.

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

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(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

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-six 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 meteorological sector in the general area of the SITE BOUNDARY; An outer ring of stations, one in each meteorological sector in the 6 mile 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.

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 five locations Three samples from close to the three SITE BOUNDARY locations, in different sectors; One sample from the vicinity of a community having the highest calculated annual average groundlevel D/Q; and One sample from a control location, as for example a population center 10 to 20 miles distant and in the least prevalent wind direction.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;[3] and gamma isotopic analysis[4] of composite (by location) quarterly.
3. Waterborne			
a. Surface[5]	One sample upstream One sample downstream	Composite sample over 1-month period.[6]	Gamma isotopic analysis[4] monthly. Composite for tritium analysis quarterly.

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 (Continued)			
b. Drinking	Two samples at each of one to three of the nearest water treatment plants that could be affected by its discharge. Two samples at a control location.	Composite sample of river water near intake at each water treatment plant over 2-week period[6] when I-131 analysis is performed, monthly composite otherwise; and grab sample of finished water at each water treatment plant every 2 weeks or monthly, as appropriate.	I-131 analysis on each sample when the dose calculated for the consumption of the water is greater than 1 mrem per year[7]. Composite for gross beta and gamma isotopic analyses[4] monthly. Composite for tritium analysis quarterly.
c. Sediment from Shoreline	One sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis[4] semiannually.
4. Ingestion			
a. Milk	Samples from milking animals in three locations within 3 miles distance having the highest dose potential. If there are none, then one sample from milking animals[8] in each of three areas between 3 and 5 miles distance where doses are calculated to be greater than 1 mrem per yr.[7]	Semimonthly.	Gamma isotopic[4,9] analysis semimonthly.

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 (Continued)			
a. Milk	One sample from milking animals[8] at a control location about 10 miles distant or beyond and preferably in a wind direction of lower prevalence.		
b. Fish	At least one sample of any commercially and recreationally important species in vicinity of plant discharge area.	Semiannually.	Gamma isotopic analysis[4] on edible portions.
	At least one sample of any species in areas not influenced by plant discharge		
	At least one sample of any anadromous species in vicinity of plant discharge.	During spring spawning season.	Gamma isotopic analyses[4] on edible portion.
c. Grass or Leafy Vegetation	One sample from two onsite locations near the SITE BOUNDARY in different sectors.	Monthly during growing season.	Gamma isotopic[4,9]
	One sample from a control location at about 15 miles distance.	Monthly during growing season.	Gamma isotopic[4,9]

TABLE 3.12-1 (Continued)

TABLE NOTATIONS

(1) Specific parameters of distance and direction sector from a point midway between the center of the two reactors, 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 ODCM. Each sample location will be designated by a number, name or some other label. 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 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 Surveillance Report pursuant to Specification 6.8.1.3. 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 alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions, if available, will be made within 30 days in the Radiological Environmental Monitoring Program given in the ODCM. Pursuant to Specification 6.13, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s), if any, with supporting information identifying the cause of the unavailability of samples for the pathway and justifying the selection of the new location(s) for obtaining samples, or the unavailability of suitable new locations.

(2) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. 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.

TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

(4) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(5) The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone.

(6) Composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

(7) The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

(8) A milking animal is a cow or goat producing milk for human consumption.

(9) If gamma isotopic analysis is not sensitive enough to meet the Lower Limit of Detection for I-131, a separate analysis for I-131 will be performed.

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)	GRASS OR LEAFY VEGETATION (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-95	400				
Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-140	200			200	
La-140	100			300	

*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.

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS(1) (2)

LOWER LIMIT OF DETECTION (LLD)(3)

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/l)	GRASS OR LEAFY VEGETATION (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58	15		130			
Co-60	15		130			
Zn-65	30		260			
Zr-95	30					
Nb-95	15					
I-131	1**	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-140	60			60		
La-140	15			15		

*If no drinking water pathway exists, a value of 3000 pCi/l may be used.

**If no drinking water pathway exists, a value of 15 pCi/l may be used.

TABLE NOTATIONS

(1)This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable as plant effluents, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Surveillance Report pursuant to Specification 6.8.1.3.

(2)Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

(3)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 system 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:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (picoCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22 = the number of disintegrations per minute per picoCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between sample collection, or end of the sample collection period, and time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

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 Surveillance Report pursuant to Specification 6.8.1.3.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A Land Use Census shall be conducted and shall identify within a distance of 5 miles the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence, and the nearest garden of greater than 500 ft^2 producing broad leaf vegetation. Land within the Savannah River Plant may be excluded from this survey.

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, pursuant to Specification 6.8.1.4, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report.

b. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM, if samples are available. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may then be deleted from this monitoring program. Pursuant to Specification 6.13, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM 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

4.12.2 The Land Use Census shall be conducted during the growing season at least once per 12 months using that information that will provide good results, such as by a door-to-door survey, by visual survey from automobile or aircraft, by consulting local agriculture authorities, or by some combination of these methods as feasible. The results of the Land Use Census shall be included in the Annual Radiological Environmental Surveillance Report pursuant to Specification 6.8.1.3.

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 that has been approved by the Commission, that correspond to samples and analysis 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 Surveillance Report pursuant to Specification 6.8.1.3.

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

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Surveillance Report pursuant to Specification 6.8.1.3.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Sections 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

3/4.0 APPLICABILITY

BASES

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 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.

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 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 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 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 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 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 ensure 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.

BASES

4.0.1 This specification provides that surveillance activities necessary to ensure 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 three 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 these 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. Items may be determined inoperable during use, during surveillance tests, or in accordance with this specification.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition 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 Condition 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.

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 1 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.

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 total loss of SHUTDOWN MARGIN in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. In MODES 1 and 2, 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 1.3% Δk/k is 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. In MODES 3, 4 and 5, the most restrictive condition occurs at BOL, associated with a boron dilution accident. In the analysis of this accident, a minimum SHUTDOWN MARGIN as defined in Specification 3/4.1.1.2 is required to allow the operator 15 minutes from the initiation of the Source Range High Flux at Shutdown Alarm to total loss of SHUTDOWN MARGIN. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting requirement and is consistent with the FSAR accident analysis assumptions. The required SHUTDOWN MARGIN is plotted as a function of RCS boron concentration.

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 condition assumed in the FSAR accident and transient analyses.

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

MODERATOR TEMPERATURE COEFFICIENT (Continued)

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 -4.0×10^{-4} $\Delta k/k/^{\circ}F$. The MTC value of -3.1×10^{-4} $\Delta k/k/^{\circ}F$ 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 -4.0×10^{-4} $\Delta k/k/^{\circ}F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the 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 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

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, and (4) the boric acid transfer pumps.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure functional capability in the event an assumed single failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions as defined by Specification 3/4.1.1.1 (MODES 1 and 2) and Specification 3/4.1.1.2 (MODES 3 and 4) 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 31740 gallons usable volume of 7000 ppm borated water from the boric acid storage tanks or 87720 gallons usable volume of 2000 ppm borated water from the refueling water storage tank (RWST).

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 changes in the event the single Boron Injection System becomes inoperable.

The boration capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 (MODE 5) after xenon decay and cooldown from 200°F to 140°F. This condition requires either 4570 gallons usable volume of 7000 ppm borated water from the boric acid storage tanks or 12630 gallons usable volume of 2000 ppm borated water from the RWST.

The contained water volume limits provided in Specifications 3/4.1.2.5 and 3/4.1.2.6 include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section are necessary to ensure that the following requirements are met at all times during normal operation. By observing that the RCCAs are positioned above their respective insertion limits during normal operation,

1. At any time in life for MODE 1 and 2 operation, the minimum SHUTDOWN MARGIN will be maintained. For operational MODES 3, 4, 5, and 6, the reactivity condition consistent with other specifications will be maintained with all RCCAs fully inserted by observing that the boron concentration is always greater than an appropriate minimum value.

2. During normal operation the enthalpy rise hot channel factor, $F_{\Delta H}$, will be maintained within acceptable limits.

MOVABLE CONTROL ASSEMBLIES (Continued)

3. The consequences of an ejected RCCA accident will be restricted below the limiting consequences referred to in the ejected rod analysis.

4. The core can be made subcritical by the required SHUTDOWN MARGIN with one RCCA stuck. In the event of an RCCA ejection, the core can be made subcritical with two RCCAs stuck, where one of the RCCAs is assumed to be the worst ejected rod control assembly.

5. The trip reactivity assumed in the accident analysis will be available.

6. Dropping an RCCA into the core or statically misaligning an RCCA during normal operation will not violate the thermal design basis with respect to DNBR.

7. The uncontrolled withdrawal of an RCCA will result in consequences no more severe than presented in the accident analysis.

8. The uncontrolled withdrawal of a control assembly bank will not result in a peak power density that exceeds the center line melting criterion.

OPERABILITY of the control rod position indicator channels (LCO 3.1.3.2) is required to determine control rod positions and thereby ensure compliance with the control rod alignment.

OPERABILITY of the Demand Position Indication System (LCO 3.1.3.2) is required to determine bank demand positions and thereby ensure compliance with the insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that some of the original criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER, either of these restrictions provide assurance of fuel rod integrity during continued operation provided no further abnormal condition develops.

For Specification 3.1.3.1 ACTIONS b and c it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus fall under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately four hours for this verification.

The maximum rod drop time permitted by (LCO 3.1.3.4) is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg}

MOVABLE CONTROL ASSEMBLIES (Continued)

551 degrees-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.

Bank demand 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.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (2) 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 elevation 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 2.30 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 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.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (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 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two 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 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

The limits on heat flux hot channel factor 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 ensure that the limits are maintained provided:

a. 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;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

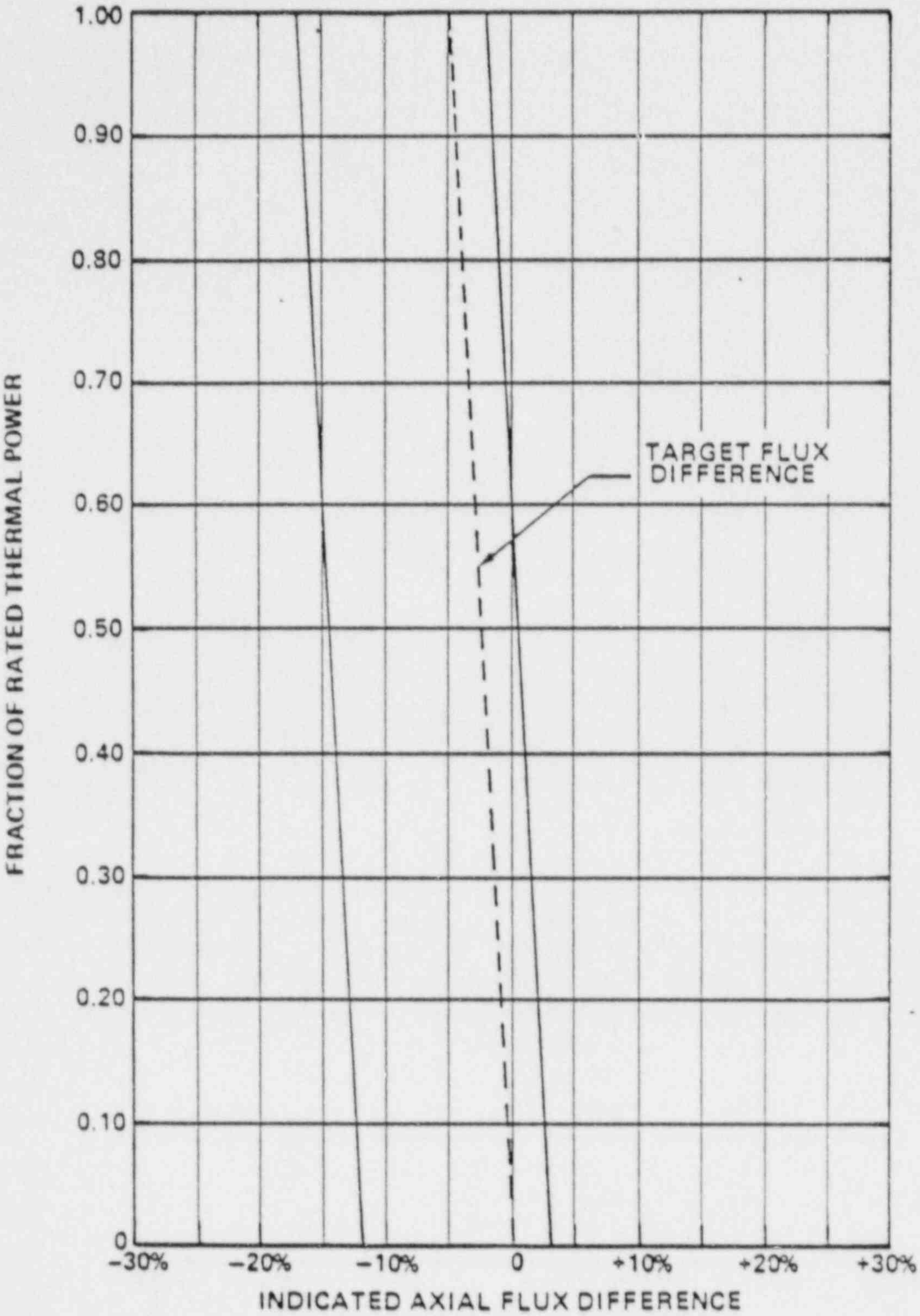


FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. 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 a. through d. above are maintained. 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.

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 $F^N_{\Delta H}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. This margin includes the following:

a. Design limit DNBR of 1.30 vs 1.28,

b. Grid Spacing (K_s) of 0.046 vs 0.059,

c. Thermal Diffusion Coefficient of 0.038 vs 0.059,

d. DNBR Multiplier of 0.86 vs 0.88, and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

BASES

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide 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 6.8.1.6 was determined from expected power control manuevers 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. A 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 control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

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 indicated T_{avg} value of 591°F and the indicated pressurizer pressure value of 2224 psig correspond to analytical limits of 592.5°F and 2205 psig respectively, with allowance for measurement uncertainty.

3/4.2.5 DNB PARAMETERS (Continued)

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. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate degradation on a 12 hour basis. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.

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 the Engineered Safety Features Actuation System instrumentation and interlocks ensures: (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 (3) 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 (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 safety 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 upon maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. The NRC Safety Evaluation Report for WCAP-10271 was provided in a letter dated February 21, 1985, from C. O. Thomas (NRC) to J. J. Sheppard (WOG-GP)

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-3 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-3. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

specified in Table 3.3-3, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, R or Rack Error is the "as measured" deviation, in the percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-3, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. Response time limits for the Reactor Trip System and Engineered Safety Features Actuation System are maintained in Table 16.3-1 and 16.3-2 of the Vogtle Electric Generating Plant Final Safety Analysis Report (VEGP FSAR). No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response time.

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) ECCS pumps start and automatic valves position, (2) Reactor trip, (3) feed water isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position (6) containment isolation, REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

(7) steam line isolation, (8) turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment fan coolers start and automatic valves position, (11) Nuclear Service Cooling and Component Cooling water pumps start and automatic valves position, and (12) Control Room Ventilation Emergency Actuation 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.

P-11 With pressurizer pressure below the P-11 setpoint, allows manual block of safety injection actuation on low pressurizer pressure signal. Allows manual block of safety injection actuation and steam line isolation on low compensated steam line pressure signal and allows steam line isolation on high steam line negative pressure rate. With pressurizer pressure above the P-11 setpoint, defeats manual block of safety injection actuation on low pressurizer pressure and safety injection and steam line isolation on low steam line pressure and defeats steam line isolation on high steam line negative pressure rate.

P-14 On increasing steam generator water level, P-14 automatically trips all feedwater isolation valves, initiates a turbine trip, and inhibits feedwater control valve modulation.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations 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 and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

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 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.

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 Guide 1.12, "Instrumentation for Earthquakes," April 1974.~~ The instrumentation on Unit 1 is shared with Unit 2 and the seismic instrumentation and corresponding Technical Specifications meet the recommendations of Regulatory Guide 1.12, Revision 1, April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are 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 SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown 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.

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, REMOTE SHUTDOWN SYSTEM (Continued)

control, and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, and control circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 and CMEB 9.5.1.

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. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," Revision 2, December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. The instrumentation listed in Table 3.3-8 corresponds to the Category 1 instrumentation for which selection, design, qualification and display criteria are described in Regulatory Guide 1.97, Rev. 2.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection Systems 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, Revision 1, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," January 1977.

This capability will not be required if the quantity of chlorine gas stored on site is small (< 20 lbs.) and utilized for laboratory and calibration purposes. This applicability is consistent with the exclusions and recommendations of Regulatory Guide 1.95, Revision 1, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," January 1977.

3/4.3.3.8 LOOSE PARTS DETECTION SYSTEM

Not used.

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 methodology and parameters in the ODCM 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 methodology and parameters in the ODCM 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 GASEOUS WASTE PROCESSING 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^{-6} µCi/ml are measurable.

3/4.3.3.11 HIGH ENERGY LINE BREAK ISOLATION SENSORS

The operability of the high energy line break isolation sensors ensures that the capability is available to promptly detect and initiate protective action in the event of a line break. This capability is required to prevent damage to safety-related systems and structures in the auxiliary building.

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.

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 MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

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 capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers.

In MODE 4, and in 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 trains/loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR trains be OPERABLE. The locking closed of the required valves in Mode 5 (with the loops not filled) precludes the possibility of uncontrolled boron dilution of the filled portion of the Reactor Coolant System. This action prevents flow to the RCS of unborated water by closing flowpaths from sources of unborated water. These limitations are consistent with the initial conditions assumed for the boron dilution accident in the safety analysis.

The operation of one reactor coolant pump (RCP) 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 an RCP with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant 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 restricting starting 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 the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR train, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Cold Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit 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 Reactor trip until the first Reactor Trip System Trip Setpoint 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.

During shutdown conditions in Mode 5 only one pressurizer code safety is required for overpressure protection. In lieu of an actual operable code safety valve an unisolated and unsealed vent pathway (i.e. a direct unimpaired opening) of equivalent size can be taken credit for an synonymous with an OPERABLE code safety.

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 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 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 Reactor Coolant System 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 PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve 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 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 Reactor Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-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 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.

STEAM GENERATORS (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission in a Special Report pursuant to Specification 6.8.2 within 30 days and 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.

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 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

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.

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 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 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.

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 64 gpm at a nominal RCS

OPERATIONAL LEAKAGE (Continued)

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 safety analyses.

The leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series 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, 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 valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an

SPECIFIC ACTIVITY (Continued)

appropriately small fraction of 10 CFR Part 100 dose guideline values 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 limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Vogtle 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 reactor 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.

The sample analysis for determining the gross specific activity and $\bar{E}$ 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 14 minutes from these determinations has been made for several 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. The choice of 14 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. 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.

SPECIFIC ACTIVITY (Continued)

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 specific count should be made for gases (i.e., xenons and kryptons) and particulates (i.e., cobalt and cesiums) 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.

The identification of 95% of the gross specific activity by definition does not obligate VEGP into calculating $\bar{E}$ every time gross activity is determined.

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 reactor 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 reactor coolant will be detected in sufficient time to take corrective action. 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

TO BE REVISED TO REFLECT UNIT 2

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 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

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/h and 200°F/h, respectively. The auxiliary spray shall not be used if the temperature difference between the pressurizer and the auxiliary spray fluid is greater than 625°F, and

5. System preservice hydrotests and inservice 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 E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 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.

The heatup and cooldown limit curves in Figures 3.4-2 and 3.4-3 are applicable to Vogtle-Unit 1 for up to 16 EFPY. The most limiting material for Vogtle-Unit 1 has an initial RT_{NDT} of 30°F and a copper content of 0.06 WT %, whereas the heatup and cooldown curves in Figures 3.4-2 and 3.4-3 are based on an initial RT_{NDT} of 40°F and a copper content of 0.10 WT %. --

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 16 effective full power years (EFPY) of service life. The 16 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 3/4.4-1. 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 phosphorus 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

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	ASME MATERIAL TYPE	CU (%)	Ni (%)	P (%)	NDTT (°F)	50 FT-LB 35 MIL TEMP (°F)	RT_{NDT} (°F)	AVG. UPPER SHELF ENERGY NMWD* (FT-LB)	MWD** (FT-LB)
Closure Head Dome	B8807-1	A533BCL.1	.16	.67	.008	-50	75	15	88	-
Closure Head Torus	B8808-1	A533BCL.1	.14	.56	.010	-30	68	8	85	-
Closure Head Flange	B8801-1	A508CL.2	-	.70	.011	20	<40	20	132	-
Vessel Flange	B8802-1	A508CL.2	-	.71	.014	0	<60	0	119	-
Inlet Nozzle	B8809-1	A508CL.2	-	.86	.011	-20	<10	-20	107	-
Inlet Nozzle	B8809-2	A508CL.2	-	.84	.014	-10	<50	-10	95	-
Inlet Nozzle	B8809-3	A508CL.2	-	.82	.013	-10	<10	-10	117	-
Inlet Nozzle	B8809-4	A508CL.2	-	.87	.014	-20	<10	-20	105	-
Outlet Nozzle	B8810-1	A508CL.2	-	.82	.006	-10	<50	-10	>124	-
Outlet Nozzle	B8810-2	A508CL.2	-	.79	.006	-10	<50	-10	>100	-
Outlet Nozzle	B8810-3	A508CL.2	-	.77	.006	-10	<50	-10	>102	-
Outlet Nozzle	B8810-4	A508CL.2	-	.80	.006	-10	<10	-10	>75	-
Nozzle Shell	B8804-1	A533BCL.1	.14	.62	.011	-10	88	28	94	-
Nozzle Shell	B8804-2	A533BCL.1	.10	.58	.006	-40	75	15	104	-
Nozzle Shell	B8804-3	A533BCL.1	.14	.69	.013	-30	100	40	92	-
Inter. Shell	B8805-1	A533BCL.1	.08	.59	.004	0	60	0	90	-
Inter. Shell	B8805-2	A533BCL.1	.08	.59	.004	-10	80	20	100	-
Inter. Shell	B8805-3	A533BCL.1	.06	.60	.003	-20	90	30	107	-
Lower Shell	B8606-1	A533BCL.1	.05	.59	.005	-50	80	20	116	-
Lower Shell	B8606-2	A533BCL.1	.05	.58	.009	-10	80	20	113	-
Lower Shell	B8606-3	A533BCL.1	.06	.64	.007	-20	70	10	118	-
Bottom Head Torus	B8813-1	A533BCL.1	.13	.50	.009	-40	50	-10	88	-
Bottom Head Dome	B8812-1	A533BCL.1	.10	.53	.009	-40	32	-28	122	-
Inter & Lower Shell Vertical Weld Seams and Girth Seam	G1.43	SAW	.03	.10	.007	-80	<-20	-80	>129	-

TO BE REVISED TO REFLECT UNIT 2

*Normal to major working directions

**Major working direction

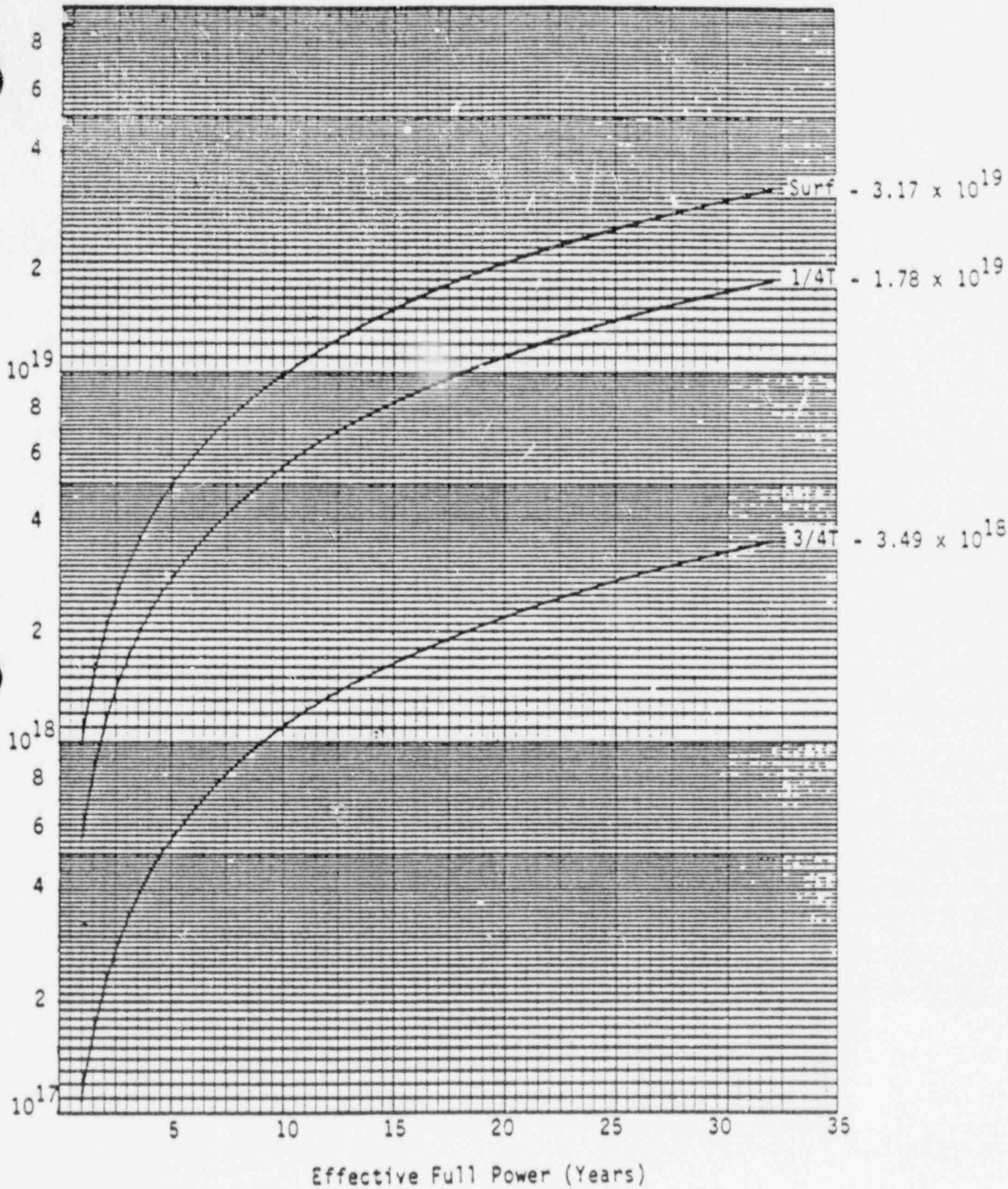
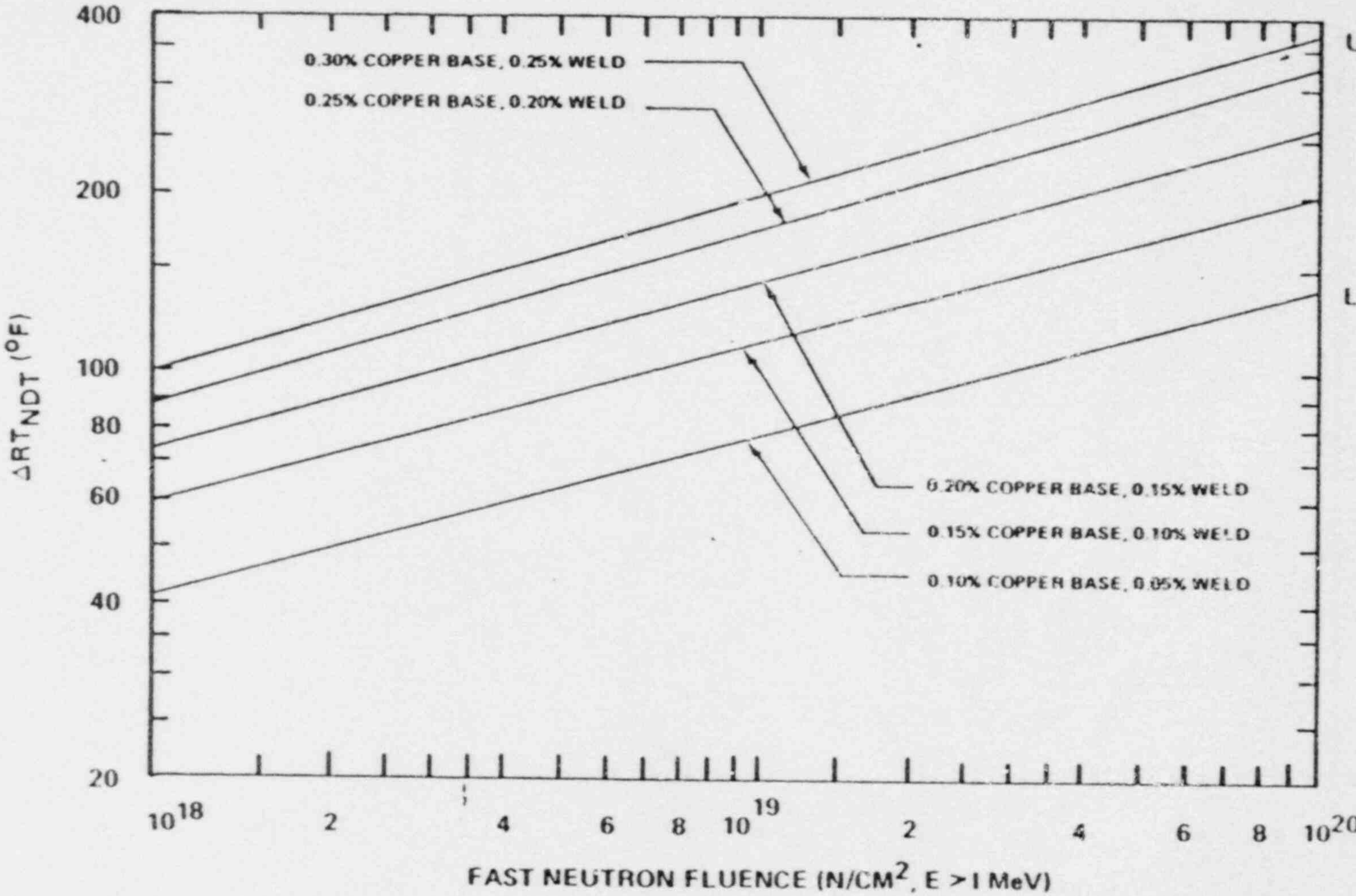


FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE



TO BE REVISED TO REFLECT UNIT 2

FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT}
FOR REACTOR VESSELS EXPOSED TO RADIATION AT 550°F

PRESSURE/TEMPERATURE LIMITS (Continued)

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 16 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{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 E185-82 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 16.3-3 of the VEGP FSAR. 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. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated 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 semielliptical 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 nonductile failure. To assure that the radiation embrittlement effects are accounted for in the 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

PRESSURE/TEMPERATURE LIMITS (Continued)

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} i- 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,

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 at 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

PRESSURE/TEMPERATURE LIMITS (Continued)

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 ΔT 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.

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

PRESSURE/TEMPERATURE LIMITS (Continued)

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.

Next, 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.

Finally, the new 10CFR50 Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. This rule states that the minimum metal temperature of the closure flange regions should be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Westinghouse Plants). For Vogtle Unit 1 the minimum temperature of the closure flange and vessel flange regions is 140°F, since the limiting RT_{NDT} is 20°F (see Table B 3/4-4.1). The Vogtle Unit 1 heatup curve shown on Figure 3-4.2 is not impacted by the new 10CFR50 rule. However, the Vogtle Unit 1 cooldown curve shown in Figure 3-4.3 is impacted by the new 10CFR50 rule.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

COLD OVERPRESSURE PROTECTION SYSTEMS

The OPERABILITY of two PORVs, two RHR suction relief valves or an RCS vent capable of relieving at least 670 gpm water flow at 470 psig 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 350°F. Either PORV or either RHR suction relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of all three charging pumps and subsequent injection into a water-solid RCS.

COLD OVERPRESSURE PROTECTION SYSTEMS (Continued)

The Maximum Allowed PORV Setpoint for the Cold Overpressure Protection System (COPS) is derived by analysis which models the performance of the COPS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that the nominal 16 EFPY Appendix G reactor vessel NDT limits criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all safety injection pumps while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The Maximum Allowed PORV Setpoint for the COPS will be updated 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 16.3-3 of the VEGP FSAR.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for 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. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through the 1975 Winter Addenda.

3/4.4.11 REACTOR COOLANT SYSTEM 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 at least one Reactor Coolant System vent path from the reactor vessel head, ensures that 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 of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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 independent 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 limitation for all safety injection pumps to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

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, (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses and (4) to ensure that centrifugal charging pump injection flow which is directed through the seal injection path is less than or equal to the amount assumed in the safety analysis.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) 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 cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

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, 2) the reactor will remain subcritical in the cold condition following a small LOCA or steamline break, assuming complete mixing of the RWST, RCS, and ECCS water volumes with all control rods inserted except the most reactive control assembly (ARI-1), and 3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow $\geq$ 3.0 FT^2) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump, post-LOCA with all control rods assumed to be out.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.6.1 PRIMARY CONTAINMENT

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 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 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 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 provides 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 psig, and (2) the containment peak pressure does not exceed the design pressure of 52 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 41.9 psig assuming an initial containment pressure of 0.3 psig. The initial positive containment pressure will limit the total pressure to less than P_a, which is less than design pressure and is consistent with the safety analyses.

Unit 1 and Unit 2 containments satisfy the recommendations of Regulatory Guide 1.35, Rev. 2, Position C1.3. Therefore, Unit 2 containment is subject to visual inspection only.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

for Unit 1, and

This limitation ensures that the structural integrity of each ~~the~~ containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 41.9 psig in the event of a steam line break accident. The measurement of containment tendon lift-off force, the and tensile tests of the tendon strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability. (The tendon strand samples will also be subjected to stress cycling tests and to accelerated corrosion tests to simulate the tendon's operating conditions and environment.)

for both units

The Surveillance Requirements for demonstrating the ~~containment's~~ structural integrity ~~are~~ of each containment is in compliance with the recommendations of Revision 2 of Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 24-inch containment purge supply and exhaust isolation valves are required to be sealed closed during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4. Sealed closed isolation valves are isolation valves under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, the use of blind flanges, or removal of power to the valve operator.

BASES

CONTAINMENT VENTILATION SYSTEM (Continued)

The use of the containment purge lines is restricted to the 14-inch purge supply and exhaust isolation valves since, unlike the 24-inch valves, the 14-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline of 10 CFR Part 100 would not be exceeded in the event of an accident during containment PURGING operation. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, should be used to justify the opening of these isolation valves.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

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 or steam line break. 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 both provide 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 volume and concentration ensure a pH value of between 8.5 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. ~~The contained solution volume limit includes an allowance for solution 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.

The solution volume limits (3700-4000 gallons) represent the required solution to be delivered (i.e., the delivered solution volume is that volume above the tank discharge).

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Cooling System 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.

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 and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. A list of containment isolation valves is provided in Table 16.3-4 of the VEGP FSAR.

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 consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," Revision 2, November 1978.

The Hydrogen Mixing Systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

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 within 110% (1304 psig) of its design pressure of 1185 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, 1974 Edition. The total relieving capacity for all valves on all of the steam lines is 18,607,220 lbs/h which is 123% of the total secondary steam flow of 15,135,453 lbs/h 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 basis:

For four loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

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 for four loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour

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-offsite power.

The auxiliary feedwater system is capable of delivering a total feedwater flow of 510 gpm at a pressure of 1221 psig to the entrance of at least two steam generators while allowing for: 1) any possible spillage through the design worst case break of the main feedwater line: 2) the design worst case single failure and, 3) recirculation flow (applicable for turbine-driven auxiliary feedwater pump only). 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 330°F, at which point the Residual Heat Removal System may be placed in operation. Because it is not desirable to inject cold auxiliary feedwater into the steam generators during power generation, the pumps must be tested on miniflow (recirculation). The surveillance acceptance criteria are based on the miniflow testing configuration and are specified to ensure the above limits are met during injection to the steam generators.

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 to maintain the RCS at HOT STANDBY conditions for 4 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power, followed by a cooldown to RHR initiation conditions. 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 offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 0.35 gpm primary-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

The identification of 95% of the gross specific activity by definition does not obligate VEGP into calculating $\bar{E}$ every time gross activity is determined.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves and bypass valves ensures that no more than one steam generator will blow down 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 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 a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the 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 NUCLEAR SERVICE COOLING WATER (NSCW) SYSTEM

The OPERABILITY of the Nuclear Service 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.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level, temperature, minimum required number of OPERABLE fans, and NSCW transfer pumps ensure that sufficient cooling capacity is available to either: (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level, maximum temperature, minimum required number of OPERABLE fans, and NSCW transfer pumps are based on providing an adequate cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," Revision 2, January 1976.

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the Control Room Emergency Air Cleanup 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 during accident conditions, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heater control circuit energized for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of

CONTROL ROOM EMERGENCY FILTRATION CLEANUP SYSTEM (Continued)

moisture on the adsorbers and HEPA filters. 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 rems 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. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.7.7 PIPING PENETRATION AREA FILTRATION AND EXHAUST SYSTEM

The OPERABILITY of the Piping Penetration Area Filtration and Exhaust System ensures that radioactive materials leaking from the containment mechanical penetration rooms and ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. Operation of the system with the heater control circuit energized 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 N510-1980 will be used as a procedural guide for surveillance testing.

3/4.7.8 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 design 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 Review Board. 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 to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. 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 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 is 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 snubbers 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 snubbers, 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.

3/4.7.9 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.

3/4.7.10 AREA TEMPERATURE MONITORING

The area temperature limitations specified in Table 3.7-3 ensure that safety-related equipment not serviced by ESF HVAC systems and necessary for safe shutdown will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY.

3/4.7.11 ENGINEERED SAFETY FEATURES (ESF) ROOM COOLER AND SAFETY-RELATED CHILLER SYSTEM

The operation of the ESF Room Cooler and Safety-Related Chiller System ensures that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment cooled by the system.

3/4.7.12 REACTOR COOLANT PUMP THERMAL BARRIER COOLING WATER ISOLATION

This isolation function is designed to prevent a spill of the reactor coolant from a postulated breached thermal barrier should a break occur in the nonsafety-related ACCW piping downstream of the isolation valve.

3/4.7.13 DIESEL GENERATOR BUILDING AND AUXILIARY FEEDWATER PUMPHOUSE ESF HVAC SYSTEMS

The operation of the diesel generator building and auxiliary feedwater pumphouse ESF HVAC systems ensures that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment served by these systems.

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

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 Criterion 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 are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of 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 the other 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 and Appendix A to Generic Letter 84-15, "Proposed Staff Position to Improve and Maintain Diesel Generator Reliability." 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 generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is 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 unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are based on the recommendations of Regulatory Guides 1.9, Revision 2 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," December, 1979; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979, Appendix A to Generic Letter 84-15 and Generic Letter 83-26, "Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests."

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the station 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 450-1975, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations," and 484-1975 "Recommended Practice for Installation Design and Installation of 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 on float 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 4.8-2 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 4.8-2 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 2.10 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

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. A list of containment penetration conductor overcurrent protective devices and feeder breakers to isolation transformers between 480 V class 1E busses and non-class 1E equipment is provided in Table 16.3-5 of the VEGP FSAR.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's 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.

The bypassing of the motor-operated valves thermal overload protection except during periodic testing ensures that the thermal overload protection will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the bypassing of the thermal overload protection continuously are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

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. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the Reactor Coolant System. This action prevents flow to the RCS of unborated water by closing flowpaths from sources of unborated water. These limitations are consistent with the initial conditions assumed for the Boron Dilution Accident in the safety analysis. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron.

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 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 BUILDING PENETRATIONS

The requirements on containment building 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 REFUELING MACHINE

The OPERABILITY requirements of the refueling machine and auxiliary hoist ensure that:

(1) The refueling machine will be used for the movement of fuel assemblies and/or rod control cluster assemblies (RCCA) or thimble plug assemblies, and the auxiliary hoist will be used for the movement of control rod drive shafts,

(2) the refueling machine will have sufficient load capacity to lift a fuel assembly and/or a rod control cluster assembly or thimble plug assembly, and the auxiliary hoist will have sufficient load capacity to lift a control rod drive shaft and attached RCCA, and

(3) the core internals and 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 AREAS

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage 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 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.

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 vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect 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 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 head removed and at least 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 vent and purge 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 STORAGE 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 POST ACCIDENT VENTILATION SYSTEM

The operability requirements on the Fuel Handling Building Post-Accident Ventilation Systems are intended to ensure that this equipment will be available in the event that a fuel handling accident results in the release of radioactive material from an irradiated fuel assembly. Although no credit is taken for the operation of this equipment in the safety analyses, its availability will serve as defense-in-depth in the event of a fuel handling accident in the fuel handling building. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

BASES

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 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 and the RCS T_{avg} may 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 to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time are 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, cannot be observed if the Position Indication Systems remain OPERABLE.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS 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 in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 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.

This specification applies to the release of radioactive materials in liquid effluents from all units at the site.

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," 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).

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 to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM 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 through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM 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

DOSE (Continued)

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 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.

3/4.11.1.3 LIQUID RADWASTE 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 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.

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.

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 and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tank's 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 and the nearest surface water supply in an UNRESTRICTED AREA.

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 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column I. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC 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 who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the whole body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

This specification applies to the release of radioactive materials in gaseous effluents from all units at the site.

The required detection capabilities for radioactive materials 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).

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 I.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

DOSE-NOBLE GASES (Continued)

MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM 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 ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

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 to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM 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 through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters 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 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 Iodine-131 Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of the 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.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the GASEOUS WASTE PROCESSING 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 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.

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 GASEOUS WASTE PROCESSING SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or 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.

3/4 11.2.6 GAS DECAY 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. Restricting the quantity of radioactivity contained in each gas decay 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.

3/4.11.3 SOLID RADIOACTIVE WASTES

This specification implements the requirements of 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, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. 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 that directly support the production of electrical power for public use exceed 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four 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 including outside storage tanks, etc.) are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 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 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 10 CFR 20.405c, 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 that is part of the nuclear fuel cycle.

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 exposure 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 measuring concentrations of radioactive materials and levels of radiation that may be compared wi' those 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., "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).

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 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 500 ft^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.

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 materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate t'at the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

SECTION 5.0

DESIGN FEATURES

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.

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 as shown in Figure 5.1-1.

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. The Site Boundary lines, plant property lines, and the exclusion area lines are shown in Figure 5.1-1. The property lines are the boundaries for determining effluent release limits.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The 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 = 226 feet.

c. Minimum thickness of concrete walls = 3 feet, 9 inches.

d. Minimum thickness of concrete roof = 3 feet, 9 inches.

e. Minimum thickness of concrete basemat = 8 feet, 3 inches.

f. Minimum thickness of concrete instrumentation cavity = 8 feet.

g. Nominal thickness of steel liner = 1/4 inches.

h. Net free volume = 2.75×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 52 psig and a temperature of 381°F.

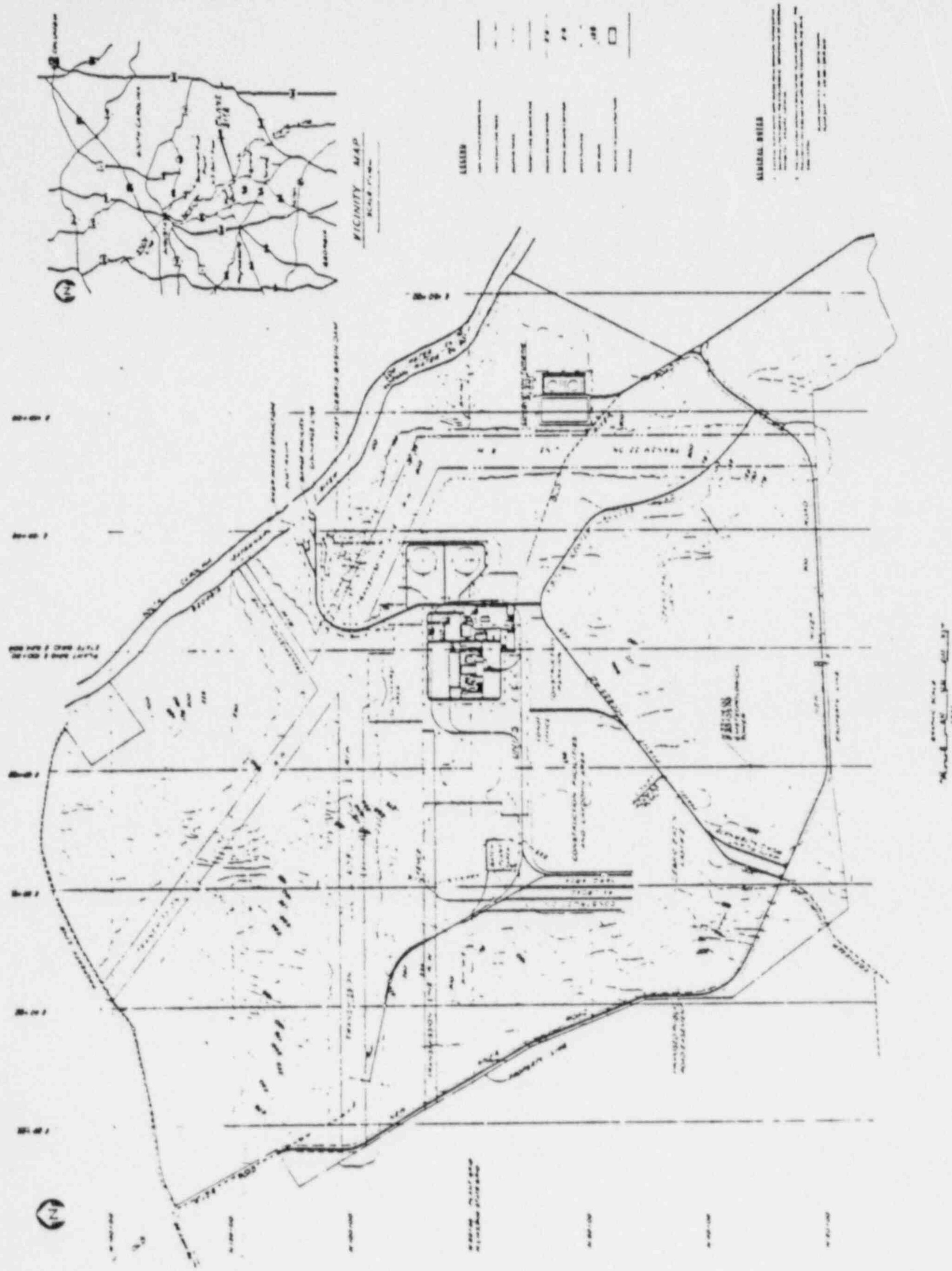


FIGURE 5.1-1

EXCLUSION AREA

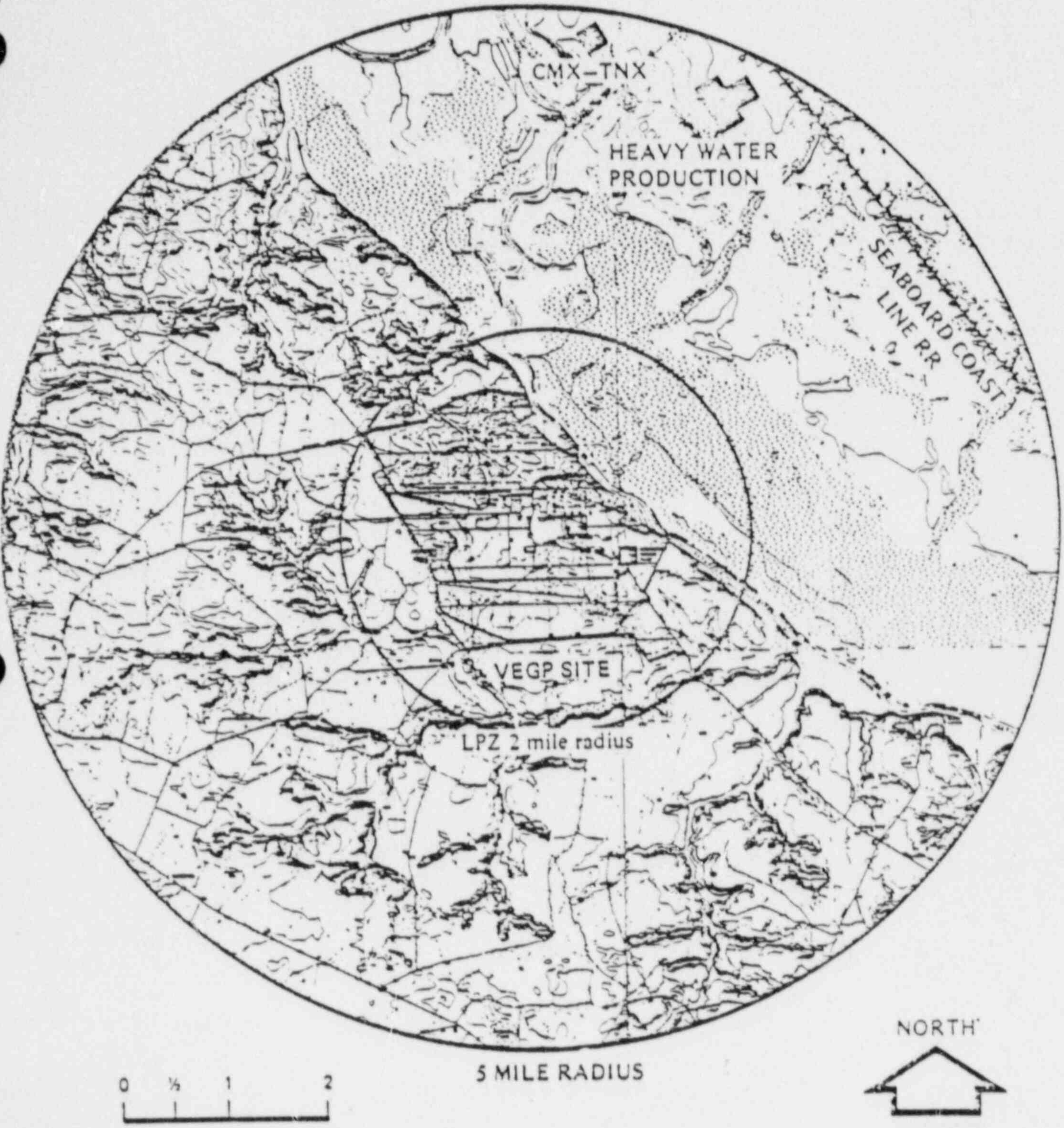


FIGURE 5.1-2

LOW POPULATION ZONE

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The 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. The initial core loading shall have a maximum enrichment not to exceed 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment not to exceed 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length control rod assemblies. The control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal absorber composition shall be 95.5% natural halfnium and 4.5% natural zirconium. All control rods shall be clad with stainless steel.

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

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,240 ± 100 cubic feet at a nominal T_{avg} of 588.5°F.

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 East 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 0.91% Δk/k for uncertainties as described in Section 4.3 of the FSAR, and

b. A nominal 10.6 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 East spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation ~~is~~ assumed.

DRAINAGE

5.6.2 The spent fuel storage pools ~~is~~ are designed and shall be maintained to prevent inadvertent draining of ~~the~~ pool below elevation 194'-1½".

CAPACITY

5.6.3 The spent fuel storage pools ~~is~~ are designed to contain sufficient storage rack locations for long-term ~~storage~~. Currently, the East pool contains two storage racks with a ~~combined capacity~~ of 288 fuel assemblies. The West pool may contain up to 20 storage racks with a combined capacity of 2098 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.

See Insert to 5-5 for 5.6.1.3 and 5.6.1.4.

5.6.1.3 The West 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 2.4% k/k for uncertainties as described in Section 4.3 of the FSAR, and

b. A nominal spacing of 10.58 inches in the North-South direction and 10.4 inches in the East-West direction between fuel assemblies placed in the storage racks.

5.6.1.4 The k_{eff} for new fuel for the first core loading stored dry in the West spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/h and 200 cooldown cycles at $\leq$ 100°F/h.	Heatup cycle - T_{avg} from $\leq$ 200°F to > 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at $\leq$ 200°F/h.	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential > 320°F and < 625°F.
	200 leak tests.	Pressurized to $\geq$ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 3107 psig.
Secondary Coolant System	1 steam line break.	Break in a > 6-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to $\geq$ 1481 psig.

6.1 RESPONSIBILITY

6.1.1 The General Manager - Vogtle Nuclear Operations (GMVNO) shall be responsible for overall plant operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The GMVNO will annually reissue a directive that emphasizes the primary management responsibility of the onshift Operations Supervisor (or during his absence from the control room, the individual designated to assume the command functions) for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for plant management and technical support shall be as shown in Figure 6.2-1.

PLANT STAFF

6.2.2 The plant organization shall be as shown in 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;

Replace with insert to page 6-1.

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. An individual* who has successfully completed the Initial Technician Training portion of the Health Physics Training Program or its equivalent shall be on site when fuel is in ~~the~~ either reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. Administrative procedures shall be developed and implemented to limit the working hours of plant staff in performance of safety-related functions (e.g., licensed Senior Operators, licensed Operators, key Health Physics Technicians, key non-licensed operators, and key maintenance personnel).

*This individual may be absent 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 position.

#- See insert to page 6-1.

Insert to page 6-1

b. When fuel is in either reactor, at least one operator licensed on the applicable unit shall be in the control room. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one Senior Operator licensed# on the applicable unit(s) shall be in the control room.

If a single Senior Operator does not hold a Senior Operator's license on both units, two or more Senior Operators who in combination are licensed as Senior Operators on both units may fulfill this requirement.

PLANT STAFF (Continued)

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the plant is operating. (This work week may consist of 12-hour shift schedules.) 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 modification, 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.

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 applicable department superintendent, 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 excess overtime shall be reviewed monthly by the General Manager - Vogtle Nuclear Operations or his designee to assure that excessive hours were authorized and that they do not become routine.

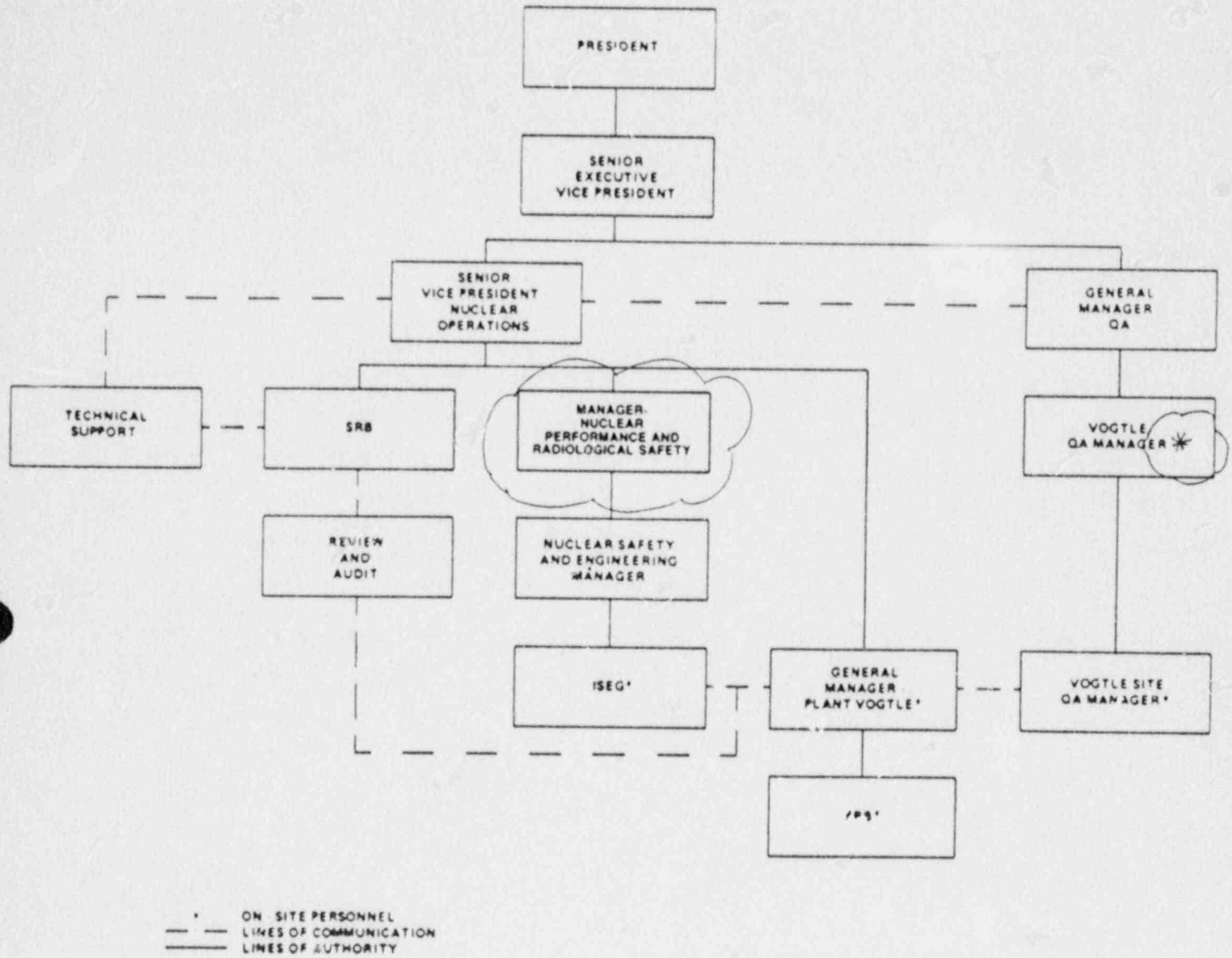


FIGURE 6.2-1

OFFSITE ORGANIZATION

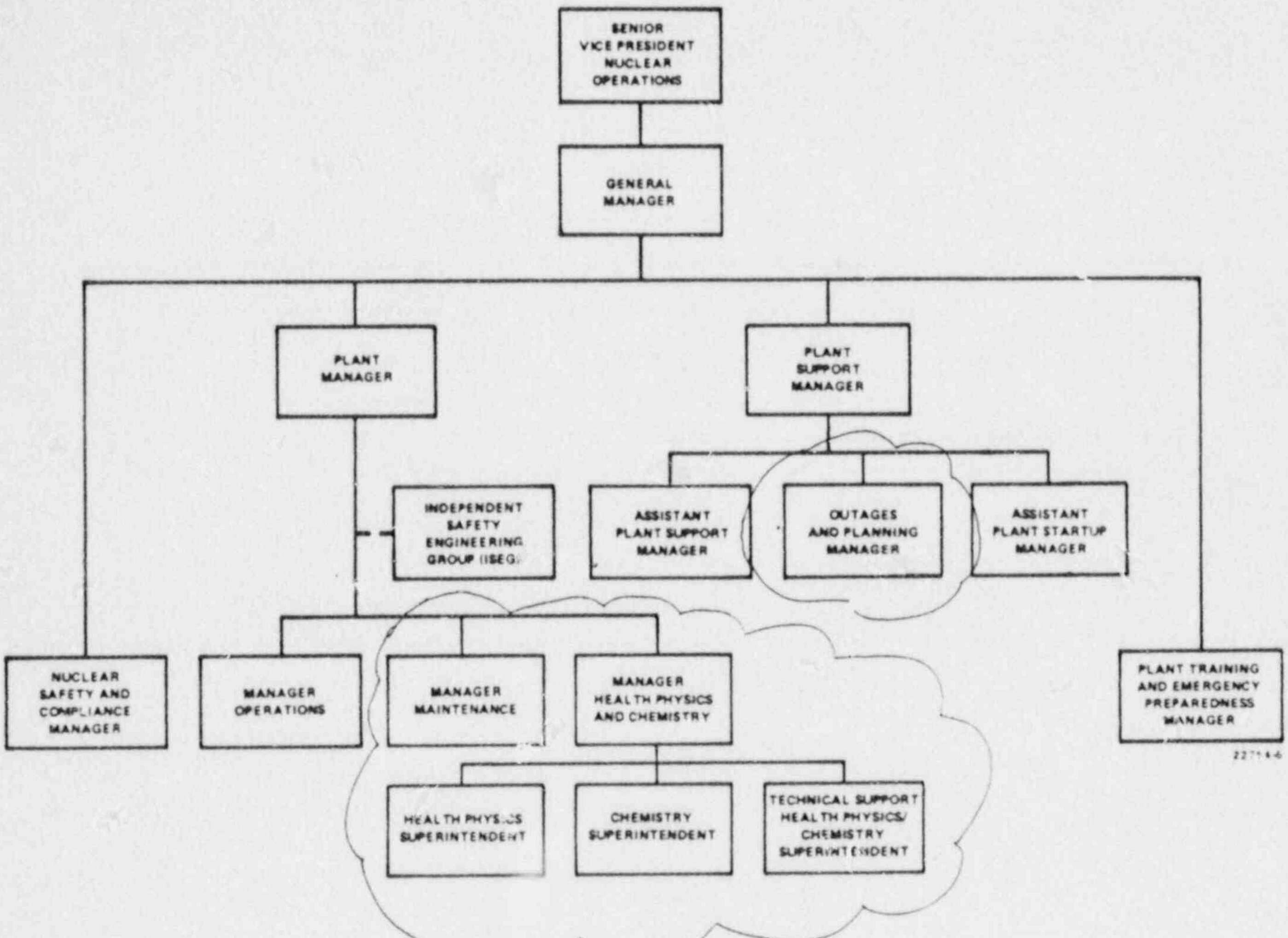


FIGURE 6.2-2

PLANT ORGANIZATION

See Insert to Page 6-5

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

~~SINGLE UNIT FACILITY~~ TWO UNITS WITH A COMMON CONTROL ROOM

~~POSITION~~	~~NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION~~	
	~~MODE 1, 2, 3, or 4~~	~~MODE 5 or 6~~
~~OS~~	~~1~~	~~1~~
~~SRO~~	~~1~~	~~None~~
~~RO~~	~~2~~	~~1~~
~~NLO~~	~~2~~	~~1~~
~~STA~~	~~1*~~	~~None~~

OS - Operations Supervisor with a Senior Operator license ~~on Unit 1~~
SRO - Individual with a Senior Operator license ~~on Unit 1~~
RO - Individual with an Operator license ~~on Unit 1~~.
NLO - Non-Licensed 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 Operations Supervisor from the control room while either ~~the~~ unit is in MODE 1, 2, 3, or 4, an individual with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Operations Supervisor from the control room while either ~~the~~ unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

* At least one of the required individuals must be assigned to the designated position for each unit.

** At least one licensed Senior Operator or Licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities must be present during CORE ALTERATIONS on either unit.

***The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Operations Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

Insert to page 6-5

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 UNITS IN MODE 1, 2, 3, or 4 AND ONE UNITS IN MODE 5 or 6 or DEFUELED
OS	1	1	1
SRO	1	none**	1
RO	3*	2*	3*
NLO	3*	3*	3*
STA	1***	none	1***

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, which may indicate areas for improving plant safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving plant safety to the Senior Vice President-Nuclear Operations through the Manager-Nuclear Performance and ~~Analysis.~~ Radiological Safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ISEG 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.

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Senior Vice President - Nuclear Operations through the Manager-Nuclear Performance and ~~Analysis.~~ Radiological Safety.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 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. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the plant for transients and accidents, and in plant design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 TRAINING

6.3.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the Plant Training and Emergency Preparedness Manager. Personnel will meet the minimum education and experience recommendations of ANSI N18.1-1971 and, for licensed staff, Appendix A of 10 CFR 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, before they are considered qualified to perform all duties independently. Prior to meeting the recommendations of ANSI N18.1-1971, personnel may be trained to perform specific tasks and will be qualified to perform those tasks independently.

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*Not responsible for sign-off function.

6.4 REVIEW AND AUDIT

6.4.1 PLANT REVIEW BOARD (PRB)

FUNCTION

6.4.1.1 The PRB shall function to advise the GMVNO on all matters related to nuclear safety.

COMPOSITION

6.4.1.2 The PRB shall be composed of Department Superintendents or Managers, or supervisory personnel reporting directly to Department Superintendents or Managers from the departments listed below:

Operations
Maintenance
Quality Control
Health Physics
Nuclear Safety and Compliance
Engineering Support

A senior health physicist is acceptable for the Health Physics Department PRB representative. The chairman, his alternate and other members and their alternates of the PRB shall be designated by the GMVNO.

ALTERNATES

6.4.1.3 No more than two alternates shall participate as voting members in PRB activities at any one time.

MEETING FREQUENCY

6.4.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman or his designated alternate.

QUORUM

6.4.1.5 The quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.4.1.6 The PRB shall be responsible for:

a. Review of 1) procedures which establish plant-wide administrative controls to implement the QA program or Technical Specifications surveillance program, 2) procedures for changing plant operating modes, 3) emergency and abnormal operating procedures, 4) procedures for effluent releases of radiological consequences, and 5) fuel handling procedures.

RESPONSIBILITIES (Continued)

b. Review of 1) programs required by Specification 6.7.4 and changes thereto, and 2) proposed procedures and changes to procedures which involve an unreviewed safety question as per 10 CFR 50.59.

c. Review of all proposed tests and experiments that affect nuclear safety;

d. Review of all proposed changes to the Technical Specifications;

e. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety;

f. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Senior Vice President-Nuclear Operations and to the Safety Review Board;

g. Review of all REPORTABLE EVENTS;

h. Review of plant operations to detect potential hazards to nuclear safety;

i. Performance of special reviews, investigations, or analyses and reports thereon as requested by the GMVNO or the Safety Review Board;

j. Review of the Security Plan and implementing procedures and submittal of recommended changes to the GMVNO and the Safety Review Board;

k. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the GMVNO and the Safety Review Board;

l. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Senior Vice President Nuclear Operations and to the Safety Review Board;

m. Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems; and

n. Review of the Fire Protection Program and Implementing procedures and submittal of recommended changes to the GMVNO.

6.4.1.7 The PRB shall:

a. Recommend in writing to the GMVNO approval or disapproval of items considered under Specification 6.4.1.6a. through e. prior to their implementation;

RESPONSIBILITIES (Continued)

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.4.1.6a. through f. constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the Senior Vice President-Nuclear Operations and the Safety Review Board of disagreement between the PRB and the GMVNO; however, the GMVNO shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.4.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Senior Vice President-Nuclear Operations and the Safety Review Board.

6.4.2 SAFETY REVIEW BOARD (SRB)

FUNCTION

6.4.2.1 The SRB 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.

The SRB shall report to and advise the Senior Vice President-Nuclear Operations on those areas of responsibility specified in Specifications 6.4.2.7 and 6.4.2.8.

COMPOSITION

6.4.2.2 The SRB shall be organized as one board for all GPC Nuclear power plants. The SRB shall be composed of a minimum of five persons who, as a group, provide the expertise to review and audit the operation of a nuclear power plant. The chairman and other members shall be appointed by the Senior Vice President-Nuclear Operations or other such person as he may designate. The composition of the SRB shall meet the requirements of ANSI N18.7-1976.

ALTERNATES

6.4.2.3 All alternate members shall be appointed in writing by the SRB Chairman to serve on a temporary basis; however, no more than a minority of alternates shall participate as voting members in SRB activities at any one time.

CONSULTANTS

6.4.2.4 Consultants shall be utilized as determined by the SRB chairman to provide expert advice to the SRB.

MEETING FREQUENCY

6.4.2.5 The SRB shall meet at least once per calendar quarter during the initial year of plant operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.4.2.6 The quorum of the SRB necessary for the performance of the SRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least a majority of the SRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the plant.

REVIEW

6.4.2.7 The SRB shall be responsible for the review of:

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~~ these Operating Licenses;

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 plant equipment that affect nuclear safety;

REVIEW (continued)

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PRB.

AUDITS

6.4.2.8 Audits of plant activities shall be performed under the cognizance of the SRB. Each inspection or audit shall be performed within the specified time interval with:

1. A maximum allowable extension not to exceed 25% of the inspection audit interval.

2. A total maximum combined interval time for any 3 consecutive inspection or audit intervals not to exceed 3.25 times the specified inspection or audit interval.

These audits shall encompass:

a. The conformance of 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 plant staff at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in plant 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 programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used as least every third year;

AUDITS (continued)

g. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months;

k. The Emergency Plan and implementing procedures (at least once per 12 months);

l. The Security Plan and implementing procedures (at least once per 12 months)~~; and~~.

RECORDS

6.4.2.9 Records of SRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each SRB meeting shall be prepared, approved, and forwarded to the Senior Vice President-Nuclear Operations within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.4.2.7 shall be prepared, approved, and forwarded to the Senior Vice President-Nuclear Operations within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.4.2.8 shall be forwarded to the Senior Executive Vice President, Senior Vice President-Nuclear Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5 REPORTABLE EVENT ACTION

6.5.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Section 50.72 and Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PRB, and the results of this review shall be submitted to the SRB and the Senior Vice President-Nuclear Operations.

6.6 SAFETY LIMIT VIOLATION

6.6.1 The following actions shall be taken in the event a Safety Limit is violated:

a. In accordance with 10 CFR 50.72, the NRC Operations Center shall be notified by telephone as soon as practical and in all cases within one hour after the violation has been determined. The Senior Vice President-Nuclear Operations, the SRB, PRB, and the GMVNO shall be notified within 24 hours.

b. A Licensee Event Report shall be prepared in accordance with 10 CFR 50.73.

c. The Licensee Event Report shall be submitted to the Commission in accordance with 10 CFR 50.73, and to the PRB, SRB, the GMVNO and the Senior Vice President-Nuclear Operations within 30 days after discovery of the event.

d. Critical operation of the affected unit shall not be resumed until authorized by the Nuclear Regulatory Commission.

6.7 PROCEDURES AND PROGRAMS

6.7.1 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. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

g. Quality Assurance for effluent and environmental monitoring;

h. Fire Protection Program Implementation; and

i. Technical Specifications Improvement Program implementation.

6.7.2 Each procedure of 6.7.1 above, and changes thereto, shall be reviewed as set forth in administrative procedures and approved by either the GMVNO or the department head of the responsible department prior to implementation with the exception of the following which shall be approved by the GMVNO:

PROCEDURES AND PROGRAMS (Continued)

1) procedures which establish plant-wide administrative controls (which implement the quality assurance program and the Technical Specifications surveillance program),

2) unit operating procedures (UOPs)

3) emergency operating procedures (EOPs)

4) abnormal operating procedures (AOPs)

5) procedures for implementing the security plan, emergency plan, and the fire protection program, and

6) fuel handling procedures.

PRB responsibilities for procedures are delineated in 6.4.1.

6.7.3 Temporary changes to procedures of Specification 6.7.1 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; and

c. The change is documented, reviewed in accordance with Specification 6.7.2 and approved by the GMVNO or department head of the responsible department within 14 days of implementation.

6.7.4 The following programs shall be established, implemented, and maintained:

a. Primary 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 the following:

1) Residual Heat Removal System
2) Containment Spray System (excluding NaOH Subsystem)
3) Safety Injection (excluding Boron Injection & Accumulators)
4) Chemical and Volume Control System (Letdown, Boron Recycle, and Charging Pumps)
5) Post Accident Processing System
6) Gaseous Waste Processing System
7) Nuclear Sampling System (Pressurizer steam and liquid sample lines, Reactor Coolant sample lines, RHR sample lines, CVCS Demineralizer and Letdown Heat Exchanger sample lines only)

PROCEDURES AND PROGRAMS (Continued)

The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

2) Leak test requirements for each system at refueling cycle intervals or less.

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,

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. Post-Accident 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.

PROCEDURES AND PROGRAMS (Continued)

e. A program which will ensure the capability to monitor plant variables and systems operating status during and following an accident. This program shall include those instruments provided to indicate system operating status and furnish information regarding the release of radioactive materials (Category 2 and 3 instrumentation as defined in Regulatory Guide 1.97 Revision 2) and provide the following:

1) preventive maintenance and periodic surveillance of instrumentation.

2) pre-planned operating procedures and back-up instrumentation to be used if one or more monitoring instruments become inoperable.

3) administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

6.8 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.8.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.8.1.1 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~~ either unit.

The initial Startup Report shall address each of the startup tests identified in the Final Safety Analysis Report 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. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications.

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 operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS**

6.8.1.2 Annual Reports covering the activities of the plant as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of plant, 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* (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). ~~The~~ The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent 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 should be assigned to specific major work functions;

b. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radioidine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

See Insert for 6-17 for footnotes ** and ***.

c. A report shall be prepared and submitted to the commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

ANNUAL RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE REPORT***

6.8.1.3 Routine Annual Radiological Environmental Surveillance Reports covering activities of the Radiological Environmental Monitoring Program 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 and shall include copies of reports of the preoperational Radiological Environmental Monitoring Program of the ~~unit~~ plant for at least two years prior to initial criticality.

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*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

NSERT FOR 6-17

** A single submittal may be made for Units 1 and 2. The submittal should combine those sections that are common to both units at the plant.

*** A single submittal may be made for Units 1 and 2.

ANNUAL RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE REPORT (Continued)

The Annual Radiological Environmental Surveillance Report~~s~~ shall include summaries, interpretations, and an analysis of trends of ~~the~~ results of the radiological environmental surveillance activities for the report period, including, as appropriate, a comparison with preoperational studies, with operational controls, and with previous environmental surveillance reports, and an assessment of any observed impacts of plant operations on the environment. The report~~s~~ shall also include the results of the Land Use Census required by Specification 3.12.2.

The Annual Radiological Environmental Surveillance Report~~s~~ shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the Offsite Dose Calculation Manual, as well as summarized and tabulated results of these analyses and measurements in the [illegible]at of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. The radiological level of radionuclides which are naturally occurring not included in the plant effluents need not be reported. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as practicable in a supplementary report.

The report~~s~~ shall also include the following: a summary description of the Radiological Environmental Monitoring Program; at least two legible maps covering all sampling locations keyed to a table giving distances and directions from a point midway between the two reactors; the results of licensee participation in the Interlaboratory Comparison Program and the corrective action taken if the specified program is not being performed as required by Specification 3.12.3; reasons 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 achieved.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.8.1.4 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.

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. For solid wastes, the format for Table 3 in Appendix B shall be supplemented with three additional categories:

SEMINANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

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 (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Report to be submitted within 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), 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~~ each 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 MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-1) 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. Historical annual average meteorological conditions or 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 MANUAL (ODCM).

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 MEMBER OF THE PUBLIC from reactor releases and other uranium fuel cycle sources within 8 km, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 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 a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), pursuant to Specifications 6.12 and 6.13, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.14. 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.

*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.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

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 Specification 3.3.3.10 or 3.3.3.11, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges 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 Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

RADIAL PEAKING FACTOR LIMIT REPORT

6.8.1.6 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be established for at least each reload core and shall be maintained available in the Control Room. The limits shall be established and implemented on a time scale consistent with normal procedural changes.

The analytical methods used to generate the F_{xy} limits shall be those previously reviewed and approved by the NRC*. If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the F_{xy} limits for all core planes containing Bank "D" control rods and all unrodded core planes along with the plot of predicted $F_Q^T \cdot P_{rel}$ vs axial core height (with the limit envelope for comparison) shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation.

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

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*WCAP 8385 "Power Distribution Control and Load Following Procedures" and WCAP 9272.A "Westinghouse Reload Safety Evaluation Methodology."

6.9 RECORD RETENTION

6.9.1 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.

6.9.2 The following records shall be retained for at least 5 years:

a. Records and logs of plant 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;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.7.1;

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.

6.9.3 The following records shall be retained for the duration of the plant Operating License:

a. Records and drawing changes reflecting plant 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 plant 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 plant staff;

RECORD RETENTION (Continued)

h. Records of inservice inspections performed pursuant to these Technical Specifications;

i. Records of quality assurance activities required by the Final Safety Analysis Report;

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 PRB and the SRB;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.8 including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality; and

n. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.10 RADIATION PROTECTION PROGRAM

6.10.1 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.

6.11 HIGH RADIATION AREA

6.11.1 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), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is greater than 100 mrem/hr but 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 a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates greater than 100 mrem/hr but 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

6.11 HIGH RADIATION AREA (Continued)

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 Health Physics Superintendent in the RWP.

6.11.2 In addition to the requirements of Specification 6.11.1, 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 RWP 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 RWP, direct or remote (such as closed circuit 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.

6.12 PROCESS CONTROL PROGRAM (PCP)

6.12.1 The PCP shall be approved by the Commission prior to implementation.

6.12.2 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

6.12 PROCESS CONTROL PROGRAM (PCP) (Continued)

3) Documentation of the fact that the change has been reviewed and found acceptable by the PRB.

b. Shall become effective upon approval by the GMVNO.

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.13.1 The ODCM shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report 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 ODCM to be changed with each page numbered, dated and containing the revision number, 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 PRB.

b. Shall become effective upon approval by the GMVNO.

6.14 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.14.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PRB. 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;

*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

6.14 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS (Continued)

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

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 a MEMBER OF THE PUBLIC 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 change is 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 PRB.

b. Shall become effective upon approval by the GMVNO.