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10 CFR 50.90

LR-N10-0088
March 23, 2010

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Salem Nuclear Generating Station, Units 1 and 2
Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

Subject: Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program

In accordance with the provisions of Title10 of the Code of Federal Regulations (10 CFR Part 50.90), "Application for Amendment of License, Construction Permit, or Early Site Permit," PSEG Nuclear LLC (PSEG) is submitting a request for an amendment to the Technical Specifications (TS) for Salem Nuclear Generating Station (Salem), Units 1 and 2.

The proposed amendment would modify Salem Technical Specifications by relocating specific surveillance frequencies to a licensee-controlled program with the implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies."

The changes are consistent with NRC-approved Industry Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specifications Task Force (RITSTF) Initiative 5b," Revision 3, (ADAMS Accession No. ML090850642). The Federal Register notice published on July 6, 2009 (74 FR 31996), announced the availability of this TS improvement.

Attachment 1 provides a description of the proposed change, the requested confirmation of applicability, and plant-specific verifications. Attachment 2 provides documentation of Probabilistic Risk Assessment (PRA) technical adequacy. Attachment 3 provides the existing TS pages marked up to show the proposed change. Attachment 4 provides the proposed TS Bases changes (for information only). Attachment 5 provides the proposed No Significant Hazards Consideration.

There are no regulatory commitments contained in this letter.

PSEG requests approval of the proposed license amendment by March 30, 2011, with the amendment being implemented within 120 days.

A001
NRR

These proposed changes have been reviewed by the Plant Operations Review Committee.

In accordance with 10 CFR 50.91, "Notice for Public Comment; State Consultation," a copy of this application, with attachments, is being provided to the designated State Official.

If you have any questions or require additional information, please contact Mr. Paul Duke at 856-339-1466.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 3/23/2010
(date)

Sincerely,

Carl J. Fricker
Site Vice President - Salem

Attachments:
1. Description and Assessment
2. Documentation of PRA Technical Adequacy
3. Proposed Technical Specification Changes
4. Proposed Technical Specification Bases Changes
5. Proposed No Significant Hazards Consideration

cc: S. Collins, Administrator, Region I, NRC
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Director - Nuclear Oversight
Director - Regulatory Affairs
Manager - Licensing
Regulatory Assurance Manager - Salem
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ATTACHMENT 1

License Amendment Request

Salem Nuclear Generating Station - Units 1 and 2
NRC Docket Nos. 50-272 and 50-311

Description and Assessment

Subject: Application for Technical Specification Change Regarding Risk -Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program

1.0 DESCRIPTION

2.0 ASSESSMENT
- 2.1 Applicability of Published Safety Evaluation
- 2.2 Optional Changes and Variations

3.0 REGULATORY ANALYSIS
- 3.1 No Significant Hazards Consideration
- 3.2 Applicable Regulatory Requirements/Criteria
- 3.3 Conclusions

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1.0 DESCRIPTION

The proposed amendment would modify the Salem Nuclear Generating Station (Salem), Units 1 and 2, Technical Specifications (TS) by relocating specific surveillance frequencies to a licensee-controlled program with the adoption of Technical Specifications Task Force (TSTF)–425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specifications Task Force (RITSTF) Initiative 5b" (Reference 1). Additionally, the change would add a new program, the Surveillance Frequency Control Program (SFCP), to TS Section 6, Administrative Controls.

The changes are consistent with NRC approved Industry/TSTF Standard Technical Specifications (STS) change TSTF-425, Revision 3, (Rev. 3) (ADAMS Accession No. ML090850642). The Federal Register notice published on July 6, 2009 (74 FR 31996) (Reference 2) announced the availability of this TS improvement.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

PSEG Nuclear LLC (PSEG) has reviewed the safety evaluation dated July 6, 2009. This review included a review of the NRC staff's evaluation, TSTF-425, Revision 3, and the requirements specified in NEI 04-10, Rev. 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies" (ADAMS Accession No. ML071360456) (Reference 3).

Attachment 2 includes PSEG's documentation with regard to Probabilistic Risk Assessment (PRA) technical adequacy consistent with the requirements of Regulatory Guide 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML070240001) (Reference 4), Section 4.2, and describes any PRA models without NRC-endorsed standards, including documentation of the quality characteristics of those models in accordance with Regulatory Guide 1.200.

PSEG has concluded that the justifications presented in the TSTF proposal and the safety evaluation prepared by the NRC staff are applicable to Salem Units 1 and 2 and justify this amendment to incorporate the changes to the Salem TS.

2.2 Optional Changes and Variations

The proposed amendment is consistent with the STS changes described in TSTF-425, Revision 3, but PSEG proposes variations or deviations from TSTF-425, as identified below.

1. Revised (clean) TS pages are not included in this amendment request given the number of TS pages affected, the straightforward nature of the proposed changes, and outstanding license amendment requests affecting some of the same TS pages. Providing only mark-ups of the proposed TS changes satisfies the requirements of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," in that the mark-ups fully describe the changes desired. This is an administrative deviation from the NRC staff's model application dated July 6, 2009 (74 FR 31996) with no impact on the NRC staff's model safety evaluation published in

the same Federal Register Notice. As a result of this deviation, the contents and numbering of the attachments for this amendment request differ from the attachments specified in the NRC staff's model application.

2. The Salem Units 1 and 2 TS were based upon NUREG-0452, Standard Technical Specifications - Westinghouse Pressurized Water Reactors, July 1979. As a result, the Salem TS Surveillance numbers differ from the Surveillance numbers in NUREG-1431 and TSTF-425, Revision 3. In addition, the Administrative Controls section of TS is Section 6.0 for Salem versus Section 5.0 for NUREG-1431. These differences are administrative deviations from TSTF-425 with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

   In addition, there are Surveillances contained in NUREG-1431 that are not contained in the Salem TS. Therefore, the NUREG-1431 mark-ups included in TSTF-425 for these Surveillances are not applicable to Salem Units 1 and 2. This is an administrative deviation from TSTF-425 with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

   The Salem TS include plant-specific Surveillances that are not contained in NUREG-1431 and, therefore, are not included in the NUREG-1431 mark-ups provided in TSTF-425. PSEG has determined that the relocation of the Frequencies for these Salem plant-specific Surveillances is consistent with TSTF-425, Revision 3, and with the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996), including the scope exclusions identified in Section 1.0, "Introduction," of the model safety evaluation, because the plant-specific Surveillances involve fixed periodic Frequencies. Changes to the Frequencies for these plant-specific Surveillances would be controlled under the SFCP. The SFCP provides the necessary administrative controls to require that Surveillances related to testing, calibration and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the Limiting Conditions for Operation will be met. Changes to Frequencies in the SFCP would be evaluated using the methodology and probabilistic risk guidelines contained in NEI 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," (ADAMS Accession No. ML071360456), as approved by NRC letter dated September 19, 2007 (ADAMS Accession No. ML072570267). The NEI 04-10, Revision 1 methodology includes qualitative considerations, risk analyses, sensitivity studies and bounding analyses, as necessary, and recommended monitoring of the performance of systems, structures and components (SSCs) for which Frequencies are changed to assure that reduced testing does not adversely impact the SSCs. In addition, the NEI 04-10, Revision 1 methodology satisfies the five key safety principles specified in Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998 (ADAMS Accession No. ML003740176) (Reference 5), relative to changes in Surveillance Frequencies. Therefore, the proposed relocation of the Salem plant-specific Surveillance Frequencies is consistent with TSTF-425 and with the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

3. The definition of STAGGERED TEST BASIS is being retained in Salem TS Definition Section 1.0 since this terminology is used in Administrative TS Sections 6.18 (Unit 1) and 6.17 (Unit 2), "Control Room Envelope Habitability Program," which is not the

subject of this amendment request and is not proposed to be changed. This is an administrative deviation from TSTF-425 with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

Additionally, Salem TS has test scheduling strategies for logic trains, channels and other components within systems that are also being relocated, consistent with the guidance of NEI 04-10, Rev 1 (Reference 3). Revision 1 to NEI 04-10 is provided to address test strategy (e.g. Staggered Test Basis) in addition to frequency. Under the proposed change, the Frequencies of all Surveillance Requirements (except those that reference other programs for the specific interval or that are event driven) are relocated. Similar to a STAGGERED TEST BASIS requirement, these Salem SRs require at least one logic train, channel or component to be tested within one interval, and all logic trains, channels or components to be tested within N intervals, where N is the total number of logic trains, channels or components subject to the test requirement. The following Surveillance Requirements (SRs) contain test scheduling requirements proposed for relocation:

- SR 4.3.1.1.3, Reactor Trip System Instrumentation Response Time
- SR 4.3.2.1.3, Engineered Safety Features Response Time
- SR 4.8.3.1a.1, 4.16 kV reactor coolant pump circuit breakers

Changes to these scheduling requirements would be controlled under the Surveillance Frequency Control Program (SFCP) which provides the necessary administrative controls for changes to testing strategies.

4. Requirements to demonstrate operability of instrument channels "during the MODES" or "for the MODES" shown in the associated TS Tables are being deleted from SRs 4.3.1.1.1, 4.3.2.1.1 and 4.3.3.1. These requirements are not restrictions on the MODES in which the SRs may be performed. The MODES shown in the associated Tables are identical to the MODES for which the instrument channels are required to be OPERABLE. SR 4.0.1 requires Surveillance Requirements to be met during the MODES or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Therefore the references to MODES in the subject SRs are redundant to the requirements of SR 4.0.1. Deleting the references to MODES from the subject SRs aligns the Salem TS more closely with NUREG-1431. This is an administrative deviation from the NRC staff's model application dated July 6, 2009 (74 FR 31996) with no impact on the NRC staff's model safety evaluation published in the same Federal Register Notice.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration

PSEG has reviewed the proposed no significant hazards consideration determination (NSHC) published in the Federal Register July 6, 2009 (74 FR 31996). PSEG has concluded that the proposed NSHC presented in the Federal Register notice is applicable to Salem Units 1 and 2 and is provided as Attachment 5 to this amendment request which satisfies the requirements of 10 CFR 50.91(a).

3.2 Applicable Regulatory Requirements/Criteria

A description of the proposed changes and their relationship to applicable regulatory requirements is provided in TSTF-425, Revision 3 (ADAMS Accession No. ML090850642) and the NRC staff's model safety evaluation published in the Notice of Availability dated July 6, 2009 (74 FR 31996). PSEG has concluded that the relationship of the proposed changes to the applicable regulatory requirements presented in the Federal Register notice is applicable to Salem Units 1 and 2.

3.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 ENVIRONMENTAL CONSIDERATION

PSEG reviewed the environmental consideration included in the NRC staff's model safety evaluation published in the Federal Register on July 6, 2009 (74 FR 31996). PSEG concluded that the staff's findings presented therein are applicable to Salem Units 1 and 2, and the determination is hereby incorporated by reference for this application.

5.0 REFERENCES

1. TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b," March 18, 2009 (ADAMS Accession No. ML090850642).

2. NRC Notice of Availability of Technical Specification Improvement to Relocate Surveillance Frequencies to Licensee Control – Risk-Informed Technical Specification Task Force (RITSTF) Initiative 5b, Technical Specification Task Force - 425, Revision 3, published on July 6, 2009 (74 FR 31996).

3. NEI 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," April 2007 (ADAMS Accession No. ML071360456).

4. Regulatory Guide 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," January 2007 (ADAMS Accession No. ML070240001).

5. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998 (ADAMS Accession No. ML003740176).

ATTACHMENT 2

License Amendment Request

Salem Nuclear Generating Station - Units 1 and 2
NRC Docket Nos. 50-272 and 50-311

PRA Technical Adequacy Assessment

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2.1 Overview

The implementation of the Surveillance Frequency Control Program (also referred to as Tech Spec Initiative 5b) at Salem will follow the guidance provided in NEI 04-10, Revision 1 [Ref. 1] in evaluating proposed surveillance test interval (STI) changes.

The following steps of the risk-informed STI revision process are common to proposed changes to all STIs within the proposed licensee-controlled program.

- Each STI revision is reviewed to determine whether there are any commitments made to the NRC that may prohibit changing the interval. If there are no related commitments, or the commitments may be changed using a commitment change process based on NRC endorsed guidance, then evaluation of the STI revision would proceed. If a commitment exists and the commitment change process does not permit the change, then the STI revision would not be implemented.

- A qualitative analysis is performed for each STI revision that involves several considerations as explained in Reference 1.

- Each STI revision is reviewed by an Expert Panel, referred to as the Integrated Decision-making Panel (IDP), which is normally the same panel as is used for Maintenance Rule implementation, but with the addition of specialists with experience in surveillance tests and system or component reliability. If the IDP approves the STI revision, the change is implemented and documented for future audits by the NRC. If the IDP does not approve the STI revision, the STI value is left unchanged.

- Performance monitoring is conducted as recommended by the IDP. In some cases, no additional monitoring may be necessary beyond that already conducted under the Maintenance Rule. The performance monitoring helps to confirm that no failure mechanisms related to the revised test interval become important enough to alter the information provided for the justification of the interval changes.

- The IDP is responsible for periodic review of performance monitoring results. If it is determined that the time interval between successive performances of a surveillance test is a factor in the unsatisfactory performances of the surveillance, the IDP returns the STI back to the previously acceptable STI.

- In addition to the above steps, the PRA is used when possible to quantify the effect of a proposed individual STI revision compared to acceptance criteria in Figure 2 of NEI 04-10. Also, the cumulative impact of all risk-informed STI revisions on all PRAs (i.e., internal events, external events and shutdown) is also compared to the risk acceptance criteria as delineated in NEI 04-10.

For those cases where the STI can not be modeled in the plant PRA (or where a particular PRA model does not exist for a given hazard group), a qualitative or bounding analysis is performed to provide justification for the acceptability of the proposed test interval change.

The NEI 04-10 methodology endorses the guidance provided in Regulatory Guide 1.200, Revision 1 [Ref. 2], "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The guidance in RG-1.200 indicates that the following steps should be followed when performing PRA assessments:

1. Identify the parts of the PRA used to support the application

   - SSCs, operational characteristics affected by the application and how these are implemented in the PRA model

   - A definition of the acceptance criteria used for the application

2. Identify the scope of risk contributors addressed by the PRA model

   - If not full scope (i.e. internal and external), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the model.

3. Summarize the risk assessment methodology used to assess the risk of the application

   - Include how the PRA model was modified to appropriately model the risk impact of the change request.

4. Demonstrate the Technical Adequacy of the PRA

   - Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.

   - Document peer review findings and observations that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.

   - Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the Regulatory Guide (currently, RG-1.200 Revision 1 includes only internal events PRA standard). Provide justification to show that where specific requirements in the standard are not adequately met, it will not unduly impact the results.

   - Identify key assumptions and approximations relevant to the results used in the decision-making process.

Because of the broad scope of potential Initiative 5b applications and the fact that the impact of such assumptions differs from application to application, each of the issues encompassed in Items 1 through 3 will be covered with the preparation of each individual PRA assessment made in support of the individual STI interval requests. The purpose of the remaining portion of this appendix is to address the requirements identified in item 4 above.

2.2 Technical Adequacy of the PRA Model

The Salem 4.3 version of the Salem PRA model is the most recent evaluation of the risk profile at Salem for internal event challenges. There is a single model used for both Salem Units 1 and 2. The Salem PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the Salem PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

PSEG Nuclear LLC (PSEG) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all PSEG nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the Salem PRA.

PRA Maintenance and Update

The PSEG risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the PSEG Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation procedures. PSEG procedure ER-AA-600-1015, "FPIE PRA Model Update" delineates the responsibilities and guidelines for updating the full power internal events PRA models at PSEG nuclear generation sites. The overall PSEG Risk Management program, including ER-AA-600-1015, defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every three years.

In addition to these activities, PSEG risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.

- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.

- Guidelines for updating the full power, internal events PRA models for PSEG nuclear generation sites.

- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65 (a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 3-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant. PSEG completed the Salem 4.3 update to the Salem PRA model in late 2009, which was the result of the incorporation of additional significant plant changes into the prior PRA model.

As indicated previously, RG-1.200 also requires that additional information be provided as part of the LAR submittal to demonstrate the technical adequacy of the PRA model used for the risk assessment. Each of these items (plant changes not yet incorporated in to the PRA model, relevant peer review findings, consistency with applicable PRA Standards, and the identification of key assumptions) will be discussed in turn.

2.2.1 Plant Changes Not Yet Incorporated into the PRA Model

A PRA updating requirements evaluation (URE) PSEG PRA model update tracking database is created for all issues that are identified that could impact the PRA model. The URE database includes the identification of those plant changes that could impact the PRA model.

As part of the PRA evaluation for each STI change request, a review of open items in the URE database for Salem will be performed and an assessment of the impact on the results of the application will be made prior to presenting the results of the risk analysis to the IDP. If a non-trivial impact is expected, then this may include the performance of additional sensitivity studies or model changes to confirm the impact on the risk analysis.

2.2.2 Applicability of Peer Review Findings and Observations

Several assessments of technical capability have been made, for the Salem PRA model. These assessments are as follows and further discussed in the paragraphs below.

An independent PRA peer review was conducted under the auspices of the PWR Owners' Group in December 2001 following the Industry PRA Peer Review process [Ref. 3]. This peer review included an assessment of the PRA model maintenance and update process.

During 2005 the Salem PRA model results were evaluated in the Westinghouse Owners Group PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process [Ref. 4].

A PRA Peer Review of the Salem PRA against Addendum B of the PRA Standard was performed during November 2008.

A summary of the disposition of the 2001 Industry PRA Peer Review facts and observations (F&Os) for the Salem PRA models was documented as part of the statement of PRA capability for MSPI in the Salem MSPI Basis Document [Ref. 5]. As noted in that document, there were no open level A or level B F&Os from the 2001 peer review.

2.2.3 Consistency with Applicable PRA Standards

As indicated above, a formal peer review was performed in November 2008. This peer review was against Addendum B of the PRA Standard [Ref. 6], the criteria in RG-1.200, Rev. 1 [Ref. 2] including the NRC positions stated in Appendix A of RG-1.200, Rev. 1 and further issue clarifications [Ref. 7]. The supporting requirements (SRs) identified from the peer review as not meeting Capability Category II that remain open are summarized in Table 2-1 along with an assessment of the impact on the base PRA.

All remaining gaps will be reviewed for consideration for the next periodic PRA model update, but are judged to have low impact on the PRA model or its ability to support a full range of PRA applications. The remaining gaps are documented in the URE database so that they can be tracked and their potential impacts accounted for in applications where appropriate.

Each item will be reviewed as part of each STI change assessment that is performed and an assessment of the impact on the results of the application will be made prior to presenting the results of the risk analysis to the IDP. If a non-trivial impact is expected, then this may include the performance of additional sensitivity studies or model changes to confirm the impact on the risk analysis.

2.2.4 Identification of Key Assumptions

The overall Initiative 5b process is a risk-informed process with the PRA model results providing one of the inputs to the IDP to determine if an STI change is warranted. The methodology recognizes that a key area of uncertainty for this application is the standby failure rate utilized in the determination of the STI extension impact. Therefore, the methodology requires the performance of selected sensitivity studies on the standby failure rate of the component(s) of interest for the STI assessment.

The results of the standby failure rate sensitivity study plus the results of any additional sensitivity studies identified during the performance of the reviews as outlined in 2.2.1 and 2.2.3 above for each STI change assessment will be documented and included in the results of the risk analysis that goes to the IDP.

Table 2-1
Identified Gaps to Capability Category II of the ASME PRA Standard

F&O	Description	Applicable SRs	Impact
IE-A1-01	A loss of an AC bus may not result in a reactor trip, but may result in a forced shutdown due to technical specifications. If the lost bus happens to be the operating bus for equipment, systems will be challenged. Loss of an AC bus is generally modeled in most PRAs. This F&O is characterized as a finding based on the lack of sufficient documentation to allow verification of SR. Include events for loss of 4Kv bus if they require a forced shutdown consistent with most industry PRAs.	IE-A1	Minimal impact, referenced event is bounded by the modeled reactor trip initiator. Initiator impact is a qualitative consideration in the NEI 04-10 methodology.
IE-A3-01	Historical events appear to lead to somewhat more complex situations than the assigned grouping would indicate. The plant-specific history indicates that on 12/31/01 an event occurred resulting in SI. The categorization of initiating events does not account for this or the case of ESFAS actuation. This F&O is characterized as a finding based on the lack of sufficient documentation to allow verification of SR. Consider re-categorizing this event as an ESF actuation (QR9).	IE-A3	Negligible impact on technical adequacy of the PRA model. Referenced event was binned as a trip with loss of Feed water, consistent with the classification scheme employed for the Salem PRA.

F&O	Description	Applicable SRs	Impact
IE-A3a-01	Although sited as an input source, there appears to be no documentation supporting a comparison of initiating events with regard to plants of similar design. The documentation indicates that "past probabilistic risk assessments" were used for source and experience. However, there is not documentation of such a comparison. It also does not identify that any examination was made for Salem-like designs. A comparison to similar designs can potentially identify those design-specific events than may have unique consequences which may not be defined in more generic sources. It also provides an industry basis for selection. This F&O is characterized as a finding based on the lack of sufficient documentation to allow verification of SR. Utilize available industry summary documentation to define generally appropriate initiating event list for specific design.	IE-A3a	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
IE-A4-01	The requirement is to address each system, including support systems to assess potential for initiating events. The analysis only addresses support systems and does not address the impact of other operating systems with regard to events resulting in a plant upset and subsequent trip signal. For charging this has the potential to impact both the initiator and response models such that consequential failures could be possible. This F&O is characterizing as a finding based on the lack of sufficient documentation to allow verification of SR. Add evaluations for frontline operating systems that in particular are part of the PRA response model.	IE-A4	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

F&O	Description	Applicable SRs	Impact
IE-A5-01	SA PRA Initiating Events Notebook, SA-PRA-001, Revision 0, Section 2.1.2 describes the review of Salem Generating Station Experience and Trip Review. No mention is made of consideration of events that occurred at conditions other than at-power operation. Also, events resulting in controlled shutdown were excluded on the basis that they present only mild challenges rather than being determined to be not applicable to at-power operation. Failure to consider non-power events and controlled shutdown events could result in exclusion of valid initiating events. Provide an explicit discussion of the review of non-power events. Improve the justification for exclusion of controlled shutdown events to address applicability to at-power operation or to provide a quantitative justification for exclusion.	IE-A5	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
IE-A6-01	SA PRA Initiating Events Notebook, SA-PRA-001, Revision 0, Section 2.1.2 does not indicate that plant operations, maintenance, engineering, and safety analysis personnel were interviewed or included in the review process for the initiating events notebook to determine if potential initiating events have been overlooked. Documentation was not available to show that the Category II/III requirement was satisfied. The initiating event analysis should document a reasonably complete identification of initiating events. Document the required interviews.	IE-A6	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

F&O	Description	Applicable SRs	Impact
IE-A7-01	SA PRA Initiating Events Notebook, SA-PRA-001, Revision 0, Section 2.1.2 does not indicate that a review of plant-specific or industry operating experience was performed for the purpose of identifying initiating event precursors. Failure to consider precursor events and controlled shutdown events could result in exclusion of valid initiating events. The model owner stated that precursors were considered during the review of plant operating experience. However, because this is not documented, the SR cannot be considered met. This should be explicitly stated in the Initiating Events Notebook.	IE-A7	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
IE-B3-01	Initiating events are not grouped with less severe events without assuming the worst potential effects. For example, the potential for a spurious SI actuation is grouped in the general transient category with events such as reactor trip and considered to be no worse than the reactor trip. However, unmitigated spurious SI events can challenge a PORV resulting in a consequential LOCA. Spurious SI events should not be grouped with general reactor trips. Also, the loss of AC power bus (F) is said to result in a degraded loss of condensate/feedwater performance. However, it is placed in the PCS available category. This presents a problem when developing the conditional failure PCS in response to the event. This F&O is characterized as a finding based on the lack of sufficient documentation to allow verification of SR. Separate out events on basis of unique impacts to the response sequence.	IE-B3, AS-A5	Minimal impact on the ability to assess impact of Surveillance interval changes. Issue only relates to development of initiating events for inclusion in the PRA model.

F&O	Description	Applicable SRs	Impact
IE-C1b-01	The loss of SW initiating event fault tree S1R4.Caf (gate IE-TSW) was reviewed and the logic appears to capture the appropriate combinations of equipment failures that contribute to the initiator. However, the documentation of the development of the initiator fault trees appears to be lacking. Section 3.3 of the Salem SA-PRA-001, Revision 0 notebook does not provide much detail of how the initiators modeled as fault trees are developed. It refers to the system model notebook. For the loss of SW initiator, the SW model notebook SA-PRA-005.13, Revision 0 was reviewed and there was no discussion of the development of the loss of SW initiator fault tree. For the loss of CC initiator fault tree, Section 4.2 of notebook SA-PRA-005.12, Revision 0 provides a good description of how that initiator fault tree is developed. This F&O is characterized as a finding based on the lack of sufficient documentation to allow verification of SR IE-C1b. Document how the loss of SW initiating event fault tree is developed. Likewise for other system initiators as needed. Include a discussion of the recoveries credited in the initiator. This should also be done for the other initiators that are fault trees.	IE-C1b, IE-C6	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
IE-C3-01	The initiators that are fault trees, such as loss of SW and loss of CC, the initiator frequency is not based on reactor year. For example, under gate IE-TSW, basic event SWS-PIP-RP-TBHDR has a mission time of 8760 hours. This F&O is characterized as a finding because it does not meet the SR. Use reactor year when quantifying the initiator frequencies.	IE-C3	This is a conservative modeling issue and would not affect the ability to assess the impact of a surveillance frequency change. If key to a specific STI assessment, could be addressed through sensitivity analyses in accordance with the NEI 04-10 methodology.

F&O	Description	Applicable SRs	Impact
AS-A7-01	Accident Sequences and Event Tree Development Notebook, SA-PRA-002, Revision 0 delineates the possible accident sequences for each modeled initiating event. However, some sequences are not explicitly modeled in the single-top fault tree (e.g., TT sequences S04 and S05 are combined into a single fault tree gate). No documentation was found to describe the basis of these combinations. Subsuming non-minimal sequences in the single-top fault tree model could result in loss of risk insights or masking of importance in non-standard configurations. Provide a description of the process used to combine non-minimal sequences with their bounding equivalent sequence in both the Accident Sequences and Event Tree Notebook or in the Quantification Notebook. Discuss how it is ensured that risk insights are not impacted by the subsuming of sequences. Provide a more complete basis for not modeling sequences judged to have "very low frequencies" such that a reviewer can evaluate the basis for the exclusion.	AS-A7	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
AS-A7-02	The VS ISLOCA sequence with no piping failure is assumed to be terminated with operator isolation of the suction path using the pump suction isolation MOVs. However, isolation cannot be accomplished until primary pressure is reduced. The potential for flooding of adjacent areas by water lost through the RHR pump seals and/or RHR heat exchangers prior to isolation does not appear to have been evaluated. Flooding of adjacent areas could impact additional equipment affecting the ability to achieve a safe, stable condition. Evaluate the potential volume of water which can be released prior to isolation of the VS sequence with no piping failure to determine if additional mitigation equipment could be affected.	AS-A7	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

F&O	Description	Applicable SRs	Impact
AS-A8-01	Accident Sequences and Event Tree Development Notebook, SA-PRA-002, Revision 0 and the associated CAFTA event trees define the end state of each sequence as success or core damage. However, the SBO sequences S08, S11, S14, and S17 are assumed to be successful based on offsite power recovery. Operator action to restore mitigating systems after power recovery is not addressed. In addition, given the fact that power recovery is only credible out to 4 hours, 20 hours of mitigating system operation and the potential failures of that equipment over a significant portion of the 24 hour mission time is not being addressed. This failure to address recovery of mitigating systems following power recovery does not ensure a safe, stable end state has been reached for some SBO sequences. There is also concern that the application of offsite power recovery is included twice in the modeling of the SBO event. Recovery is credited in the application of a diesel mission time of 6 hours and again through the application of offsite power recovery top event RBU. Recovery of offsite power does not guarantee restoration of mitigating systems needed to establish a safe stable condition in the plant. In some plant models, operator action to restore required mitigating systems following power recovery has been shown to be significant. In addition, mitigating system operation over a significant portion of the 24 mission time is not being addressed. Extend the event tree models to address restoration and operation of required safety functions following offsite power recovery. Potential events to include are decay heat removal and primary inventory makeup.	AS-A8, AS-B6	Mission times vice recovery of off site power are addressed in current PRA model. Documentation to be updated to reflect this.

F&O	Description	Applicable SRs	Impact
SC-B5-01	A check of the reasonableness and acceptability of the success criteria results is not documented. Comparing success criteria results with those of similar plants or performed using other plant-specific codes provides greater assurance that the results are correct. Document a check of the reasonableness and acceptability of the success criteria results. Supporting requirement SC-B5 provides example methods. Note that the PWROG PSA database identifies success criteria for its constituent plants and may be a helpful resource.	SC-B5	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SC-C3-02	Sources of uncertainty are addressed in a draft evaluation using guidance from draft EPRI report, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments." An appropriate characterization of uncertainty is required to support risk-informed decision making. Apply the EPRI guidance, once finalized, to identify the sources of uncertainty in the analysis.	SC-C3, AS-C3, HR-D6, HR-G9, HR-I3, DA-E3, QU-E1, QU-E3, QU-E4, QU-F4, LE-F2, LE-G4	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model. The NEI 04-10 methodology explicitly addresses uncertainty.
SY-A4-01	System walkdown documentation not included in the system notebook documentation. A review of system notebooks and available documentation does not include system walkdown information. A draft document containing photos and documentation of insights from a system walkdown was provided to the peer review but is not finalized. Finalize the provided notebook.	SY-A4	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

F&O	Description	Applicable SRs	Impact
SY-A6-01	Missing boundary definitions for system models. The system notebooks do not clearly define the boundaries. The training documentation is not adjusted to be specific to the PRA model. Additionally some systems, such as ac power, do not include discussion of modeled events. The diesel generator and the fuel oil transfer system are not addressed explicitly. Develop PRA specific illustrations and expand documentation to clearly describe the system boundaries to ensure that no components are double counted or missed.	SY-A6, SY-C2	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-A8-01	Review of notebooks and data notebook did not provide a source for inclusion or exclusion of failure modes based on data boundaries. No documentation of component boundaries. Expand the data discussion to provide component definitions.	SY-A8	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-A10-01	Some systems do not include expected failure modes and although this may be correct, there is no documentation as to how the data boundaries encompass the expected failures. One example is the diesel generator model does not include the diesel generator day tank and instrumentation. The response to inquiries was that these components are part of the diesel skid package. This is usually separate modeling to capture miscalibrations. Define what is included within the diesel generator "box" or expand the model.	SY-A10	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

F&O	Description	Applicable SRs	Impact
SY-A12-01	Review of system models identified some missing component failure modes. Required components are not always addressed in the model. For example, the diesel generator day tank and fuel oil check valves are not included. Additionally, restart of some components (such as dampers having to re-open for CAV) are absent in the model. Define boundaries to show incorporation of failure modes by other events or expand model.	SY-A12	Components are included in the PRA model either explicitly or as part of a super-component This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-A13-01	The modeling excludes some required component failures without justification. Some failure modes listed for inclusion in the SR are not found or are excluded from the model. This includes the transfer closed/plugging failure modes for valves and the absence of some check valves and/or tanks. Justify the exclusion of any failure mode or model the failure mode.	SY-A13	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-A16-01	The SWS fault tree includes recovery via alignment of the header crosstie. HFE SWS-XHE-FO-OVER2 is used for this recovery action in all cases, even LOSP. However, the timing used in the HRA for this action is based on room heatup following a loss of CAV, not on the more restrictive timing required for recovery of cooling to a diesel following LOSP. Application of the HFE for recovery of SW via the header crosstie in the incorrect context may result in underestimating the importance of the HFE and associated equipment required for the recovery. Create a variation of the SWS-XHE-FO-OVER2 HFE accounting for differences in timing during LOSP conditions where cooling to a diesel generator is required.	SY-A16	If determined to impact a specific STI analysis, a sensitivity analysis in accordance with the NEI 04-10 methodology would be performed.

F&O	Description	Applicable SRs	Impact
SY-A19-01	System notebooks do not include discussions on potential adverse operating conditions that could impact operation. No documentation of any potential for loss of desired system function, e.g., excessive heat loads, excessive electrical loads, excessive humidity, etc. Add brief discussion.	SY-A19, SY-A20	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-A21-01	The current type code does not provide consistent nomenclature for same failure data. The SR indicates that the nomenclature should use the same identifier for the same failure mode. The type code changes by system although the data is from the same source. For data sources from the same reference the same type code should be used. Using type codes by system may obscure the state of knowledge information.	SY-A21, QU-A2b	This observation is believed to relate to special instances where data is lacking for certain component/failure modes (e.g. diesel driven air compressor fails to start). In these cases data from other similar components were used. This is believed to be an appropriate approximation. A STI assessment is for specific components or systems, as such specific basic events are identified in the analysis. This issue does not affect the ability to perform a STI assessment.
SY-B5-01	Documentation indicated that the heated water circulating system was required. Documentation for several system notebooks (AFW, CVCS and RWST) indicated that the heated water circulating system was required to prevent freezing, but was not modeled. Model the heated water circulating system or justify the reason for not modeling.	SY-B5	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

F&O	Description	Applicable SRs	Impact
SY-B11-01	Some AFW signals (SI, LOSP) are not defined and no justification for exclusion is provided. The SR states that actuation signals must be considered or justification provided. The AFW start signals are not completely modeled and justifications for exclusion are not provided. Provide justification for exclusion of the AFW signals or model these signals.	SY-B11	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
HR-C3-01	There is no documentation showing that miscalibration as a mode of failure of initiation of standby systems was considered. An example of this is that there is no HFE for miscalibration of bus undervoltage bus, RPS relays, etc. There is no documentation showing that miscalibration as a mode of failure of initiation of standby systems was considered. An example of this is that there is no HFE for miscalibration of bus undervoltage bus, RPS relays, etc. Consider analyzing the miscalibration of standby systems.	HR-C3, SY-B16	If the surveillance frequency for the referenced instrumentation would be evaluated, the NEI 04-10 methodology allows for modification of the PRA model if needed to perform the STI assessment.

F&O	Description	Applicable SRs	Impact
DA-A1a-01	No discussion of component boundary definition is provided in either the data or systems analysis. Boundaries for unavailability events are not established. Boundary definitions help assure that failures are attributed to the correct component and that calculated failure rates and unavailability values are appropriate. Some component boundaries are discussed in the notes to Appendix A, "Generic (Industry) Failure Data" of the Data Notebook. Note 32 states to "Assume that CCW/RHR HX failure rates apply to TDAFW Pump Bearing and governor jacket coolers", however unless the Salem TDAFW pump has unique features that require this to be modeled separately, cooling to the TDAFW pump is usually included in the component boundary to the pump. Define the component boundary for each component consistent with the failure data source. Establish boundaries for unavailability events consistent with definitions in the systems analysis.	DA-A1a, DA-C1	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
DA-C1-01	Generic unavailability data is used for some SSCs without demonstrating that the data is consistent with the test and maintenance philosophies for the subject plant. Generic unavailability data may not be applicable to Salem if its T&M approach is different. Review and state that any generic unavailability data used is applicable to the Salem model.	DA-C1	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
DA-C2-01	Plant-specific data is only collected for MSPI components. The PRA procedure requires plant specific data to be collected for components with a RAW >2 or a F-V greater than 0.005. MSPI components are only a subset of the risk-significant components. Expand collection of plant-specific data to all modeled components or justify why the generic data is applicable.	DA-C2	The majority of high importance systems were updated with recent plant specific data. The NEI 04-10 methodology requires failure rates sensitivities as part of the analysis which will address this gap.

F&O	Description	Applicable SRs	Impact
DA-C6-01	Documentation describing the process of evaluating the number of plant specific demands for standby components could not be identified. Standby components were identified in Table 1 of the Data Analysis Notebook and plant specific demands for some of these components were listed in Appendix B, however the basis for these numbers of demands was not provided. The draft data procedure states that plant specific data should be estimated by actual counts of hours or demands from logs or counters, use of surveillance procedures to estimate the frequency of demands and run times, or estimates based upon input from the System Engineer. Failure rates are dependent on an accurate demand count or component importance could be skewed. Standby components were identified in Table 1 of the Data Analysis Notebook and plant specific demands for some of these components were listed in Appendix B, however the basis for the number of demands was not provided. The draft data procedure states that plant specific data should be estimated by actual counts of hours or demands from logs or counters, use of surveillance procedures to estimate the frequency of demands and run times, or estimates based upon input from the System Engineer. Issue the data collection guidance document and document/justify the basis for the demands used.	DA-C6	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
DA-C10-01	Compare the initiator frequencies used in the Salem model with other generic data sources. This F&O is characterized as a suggestion because the IE notebook does include a comparison with NUREG/CR-5750. But, it is recommended for completeness to check how the Salem set of initiators compares with other data sources.	DA-C10	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

F&O	Description	Applicable SRs	Impact
DA-D4-01	No documentation exists related to the comparison between the generic value and the plant-specific update value to ensure accurate and meaningful implementation of Bayes approach. The documentation only indicates that data came from NUREG/CR-6928 and that MSPI data was used to perform the update. It then references Appendix B which is only Table B-1. The table provides limited information related to the update and does not provide any comparisons of results or discussions with regard to applicability of results. The documentation only indicates that data came from NUREG/CR-6928 and that MSPI data was used to perform the update. It then references Appendix B which is only Table B-1. The table provides limited information related to the update and does not provide any comparisons of results or discussions with regard to applicability of results. Perform comparisons of results with regard to initial ranges of possible generic values and confirmation that the updated results are within the expected range. Also confirmation that plant data, due to relatively small generic alpha factors is not biasing the updated value.	DA-D4	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
DA-D6-01	No documentation is present that provides any comparisons between data sources. Perform the evaluation.	DA-D6	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
DA-E2-01	A draft document was provided that documented how to establish component boundaries, how to establish failure probabilities, sources of generic data, etc. This procedure needs to be formalized. The draft document discussing how to perform data analyses needs to be finalized to ensure quality. Provide procedure on how to perform data analysis.	DA-E2, DA-C6, DA-C7, DA-C8, DA-C9	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

F&O	Description	Applicable SRs	Impact
QU-B5-01	Salem Quantification Notebook SA PRA-2008-01 Attachment E documents the convergence analysis performed to set an appropriate truncation value. The truncation level for both CDF and LERF was set at 1.0E-11. The percentage change between 1.0E-10 and 1.0E-11 was 2.2% for CDF, but 6.1% for LERF. Therefore, this SR was not satisfied for LERF. The supporting requirement applies the same criteria for convergence to both CDF and LERF. The criteria were satisfied for CDF, but not LERF. Document the overall philosophy and method for breaking circular logic in the Quantification notebook and provide sufficient documentation in the system notebooks to provide assurance that unnecessary conservatisms or non-conservatisms are not introduced.	QU-B5	The convergence issue is addressed in the current PRA model. The circular logic issue is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
QU-B9-01	Split fractions and undeveloped events are included in the model. Examples include main feedwater availability for ATWS (MFI-UNAVAILABLE) and some Unit 2 systems credited for recovery of Unit 1 CAV failure (G2SW22). The derivation of the values for these events is not documented. The derivation of split fractions and undeveloped events is not documented sufficiently to allow identification of shared events and results interpretation based on individual events subsumed into the split fraction. Document the derivation of any split fractions and undeveloped events used in the model sufficiently to allow results interpretation and to provide assurance that the impact of any shared components is appropriately considered.	QU-B9	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

F&O	Description	Applicable SRs	Impact
QU-D1b-01	There is no discussion in the quantification notebook that indicates a review of the results was performed for the purpose of assessing modeling and operational consistency. Also, since the sequences were not quantified, it is difficult to perform this verification. This F&O is characterized as a finding because there was insufficient documentation available to verify the SR. Review the results for modeling consistency (e.g., event sequence models consistency with systems models and success criteria) and operational consistency (e.g., plant configuration, procedures, and plant-specific and industry experience) and include in the quantification notebook.	QU-D1b	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
QU-D3-01	This is a Capability Category I because there is no documentation to indicate that the Salem results were compared to the results of a similar plant. No documentation was provided showing this requirement was met. Provide a comparison of initiating event contributions and significant basic event importances between Salem and similar plants based on information available in the PWROG PRA Comparison Database.	QU-D3	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
QU-D4-01	There is no documentation indicating that a sampling of non-significant accident cutsets or sequences were reviewed to determine they are reasonable and have physical meaning. Quantification Notebook Section 2 only requires a review of the top 100 cutsets. Review of a sampling of non-significant cutsets can also reveal logic problems or recovery rules which are not being applied correctly. Include a requirement for review of a sampling of non-significant sequences in Section 2 of the Quantification Notebook and in procedures governing the model update process.	QU-D4	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
QU-F2-01	This requirement was only partially met as described below: (a) This requirement is met by the system and HRA notebooks.	QU-F2, QU-F3,	This is judged to be a documentation consideration only and does not affect

F&O	Description	Applicable SRs	Impact
	(b) There is a cutset review process description (c) There is no description of how the success systems are accounted for. Since a one top tree is used the software already accounts for this. A statement stating would be satisfactory. The truncation values and how they were determined were documented. The method for applying recovery and how post initiator HFE's are applied was not described. (d) This requirement was met. (e) This requirement was met (f) This requirement was not met since the cutsets per accident sequence were not discussed. (g) This requirement was not met since equipment or human actions that are the key factors in causing the accidents sequences to be are not discussed. (h) This requirement was not met since sensitivities were not documented. (i) This requirement was not met since the uncertainty notebook was not finalized. (j) This requirement is not met since there is no discussion of importance. (k) This requirement is not met because there is not list of mutually exclusive events and there justification. (l) This requirement is not met because there is no discussion of asymmetries in quantitative modeling to provide application users the necessary understanding regarding why such asymmetries are present in the model. (m) This requirement is met since CAFTA and Forte are being used. Both of these pieces of software are industry standards and therefore no further testing is required. Several documentation items called for in this supporting requirement were not available for review. Specific items not included in the documentation were: the process used to account for system successes, accident sequence results, discussion of factors causing accidents to be non-dominant, sensitivity assessments, uncertainty distribution,	QU-F6	the technical adequacy of the PRA model.

F&O	Description	Applicable SRs	Impact
	importance measure results, basis for elimination of mutually exclusive events, asymmetries in the model, and a quantitative definition of significant basic event, significant cutset, and significant accident sequence. Expand the documentation to address the items documented in the F&O.		
LE-C8a-01	Equipment survivability and human actions under adverse environments must be considered to reach Category II. No documentation provided or credit taken for equipment or operators in adverse environment. Provide discussion on environmental conditions and the effects on operator actions.	LE-C8a, LE-C8b, LE-C9a, LE-C9b	No credit is taken for equipment or actions under adverse environments. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
LE-F1b-01	Other than verifying that the sum of the three end states (INTACT, LATE and LERF) is approximately equal to the core damage frequency, no checks on the reasonableness of the LERF contributors is documented. A review for reasonableness is required to meet the intent of this SR. Review contributors for reasonableness (e.g., to assure excessive conservatisms have not skewed the results, level of plant specificity is appropriate for significant contributors, etc.).	LE-F1b	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
LE-F3-01	LERF uncertainties are not characterized consistent with the requirements in Tables 4.5.8-2(d) and 4.5.8-2(e). LERF uncertainties must be appropriately characterized to meet the intent of this SR. Characterize the LERF uncertainties consistent with the requirements in PRA Standard Tables 4.5.8-2(d) and 4.5.8-2(e).	LE-F3	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model. The NEI 04-10 methodology explicitly addresses uncertainty.

F&O	Description	Applicable SRs	Impact
LE-G5-01	Limitations in the LERF analysis that would impact applications are not documented. Limitations in the LERF analysis that would impact applications must be discussed to meet this SR. Document the limitations in the LERF analysis that would impact applications.	LE-G5	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
LE-G6-01	A definition for significant accident progression sequence is not documented. A definition for significant accident progression sequence must be included to meet this SR. Include in the documentation a definition for significant accident progression sequence.	LE-G6	This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

2.3 External Events Considerations

External hazards were evaluated in the Salem Individual Plant Examination for External Events (IPEEE) submittal in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of associated severe accident risks.

The results of the Salem IPEEE study are documented in the Salem IPEEE [Ref. 8]. Each of the Salem external event evaluations were reviewed as part of the submittal by the NRC and compared to the requirements of NUREG-1407. The NRC transmitted the Staff Evaluation Report of the Salem IPEEE Submittal to PSEG in 1999 [Ref. 9].

Consistent with Generic Letter 88-20, the Salem IPEEE Submittal does not screen out seismic or fire hazards, but provides quantitative analyses. The seismic risk analysis provided in the Salem Individual Plant Examination for External Events is based on a detailed Seismic Probabilistic Risk Assessment, or Seismic PRA.

The Salem Seismic PRA study was a detailed analysis that uses quantification and model elements (e.g., system fault trees, event tree structures, random failure rates, common cause failures, etc.) consistent with those employed in the internal events portion of the Salem IPE study. Salem currently does not maintain a Seismic PRA.

The internal fire events were addressed by using a combination of the Fire Induced Vulnerability Evaluation (FIVE) methodology [Ref. 10] and Fire PRA techniques. The Salem Fire PRA study is a detailed analysis that uses quantification and model elements (e.g., system fault trees, event tree structures, random failure rates, common cause failures, etc.) consistent with those employed in the internal events portion of the Salem IPE study. Salem currently does not maintain a Fire PRA.

As such, there are no comprehensive CDF and LERF values available from the IPEEE to support the STI risk assessment.

In addition to internal fires and seismic events, the Salem IPEEE analysis of high winds or tornados, external floods, transportation accidents, nearby facility accidents, release of onsite chemicals, detritus and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards.

2.3.1 Discussion of External Events Evaluations

<u>Seismic PRA</u>

The Salem IPEEE Seismic PRA was developed using a process as described in the IPEEE submittal and summarized below:

- Seismic hazard analysis
- Seismic fragility assessment

- Seismic systems analysis
- Quantification of Seismic CDF

Salem Seismic PRA methodology includes the following:

- Seismic fragilities based on revised Lawrence Livermore National Laboratory (LLNL) seismic hazard estimates. The EPRI site specific seismic hazard study are used as input as a sensitivity case.
- A seismic event is not assumed to result in a Loss of Offsite Power (LOOP). Seismic failure of offsite power is evaluated on a probabilistic basis according to component fragilities.

The Salem IPEEE states that no plant unique vulnerabilities associated with the Seismic Analysis were identified. As identified above, the seismic PRA is not currently maintained for Salem. Thus, quantitative insights can be derived based on the seismic PRA or a qualitative assessment can be performed.

Fire PRA

The Salem IPEEE Fire PRA was developed using a multi-step process as described in the IPEEE submittal and summarized below:

- Step 1 - Fire compartment interaction analysis
- Step 2 - FIVE methodology quantitative screening
- Step 3 - Develop fire PRA analysis in accordance with NUREG/CR-2300 and NUREG/CR-4840

Some of the highlights of the Salem Fire IPEEE methodology include the following:

- Fire initiation frequencies based on the FIVE methodology.
- High hazard rooms (those that contain a large amount of combustibles) were specifically analyzed.

The Salem IPEEE states that no fire induced vulnerabilities were identified as a result of the analysis. As identified above, the fire PRA is not currently maintained for Salem. Thus, quantitative insights can be derived based on the IPEEE fire PRA or a qualitative assessment can be performed.

Other External Hazards

The other external hazards are assessed to be non-significant contributors to plant risk:

- High Winds / Tornadoes: They were either screened by comparison to the SRP criteria or by bounding probabilistic evaluation. No issues were identified.

- Transportation and Nearby Facility Hazards: Transportation and nearby hazards and river traffic hazards were evaluated by screening analyses

- External Floods: They were screened by bounding probabilistic evaluation

- River detritus was evaluated via a screening analysis that included upgrades for protection of the Circulating water system that had been made.

The NEI 04-10 methodology allows for STI change evaluations to be performed in the absence of quantifiable PRA models for all external hazards. For those cases where the STI cannot be modeled in the plant PRA (or where a particular PRA model does not exist for a given hazard group), a qualitative or bounding analysis is performed to provide justification for the acceptability of the proposed test interval change.

Therefore, in performing the assessments for the other hazard groups, the qualitative or bounding approach will be utilized in most cases.

2.4 Summary

The Salem PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in risk-informed processes such as that proposed for the implementation of a Surveillance Frequency Control Program. As indicated above, in addition to the standard set of sensitivity studies required per the NEI 04-10 methodology, open items for changes at the site and remaining gaps to specific requirements in the PRA standard will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

2.5 References

[1] *Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies, Industry Guidance Document*, NEI 04-10, Revision 1, April 2007.

[2] Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities*, Revision 1, January 2007.

[3] Nuclear Energy Institute Industry PRA Peer Review Process, NEI-00-02, March 2000.

[4] *Westinghouse Owner's Group Mitigating Systems Performance Index Cross Comparison*, WCAP-16464-NP, Revision 0, July 2005.

[5] Salem MSPI Basis Document, Rev. 7, December 2009.

[6] American Society of Mechanical Engineers, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*, (ASME RA-S-2002), Addenda RA-Sa-2003, and Addenda RA-Sb-2005, August 2007.

[7] U.S. Nuclear Regulatory Commission Memorandum to Michael T. Lesar from Farouk Eltawila, "Notice of Clarification to Revision 1 of Regulatory Guide 1.200", July 27, 2007.

[8] PSEG, *Salem Generating Station Individual Plant Examination for External Events*, January 1996.

[9] U.S. Nuclear Regulatory Commission, Review of Individual Plant Examination for External Events (IPEEE) Submittal for Salem Generating Station, May 1999.

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ATTACHMENT 3

License Amendment Request

Salem Nuclear Generating Station - Units 1 and 2
NRC Docket Nos. 50-272 and 50-311

Proposed Technical Specification Changes

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INSERT 1

In accordance with the Surveillance Frequency Control Program

INSERT 3

6.8.4.l Surveillance Frequency Control Program
This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - Tavg > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥ 1.3% Δk/k.

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

ACTION:

With the SHUTDOWN MARGIN < 1.3% Δk/k, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 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 ≥ 1.3% Δk/k:

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

INSERT 1

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that control banks are within the limits in the COLR per Specification 3.1.3.5.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits in the COLR per specification 3.1.3.5.

---

* See Special Test Exception 3.10.1

\# With $K_{eff} \geq 1.0$

\## With $K_{eff} < 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit in the COLR per Specification 3.1.3.5.

INSERT 1

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

1. Reactor coolant system boron concentration,

2. Control rod position,

3. Reactor coolant system average temperature,

4. Fuel burnup based on gross thermal energy generation,

5. Xenon concentration, and

6. Samarium concentration.

INSERT 1

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 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

SHUTDOWN MARGIN - T_{avg} ≤200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be ≥ 1.0% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN < 1.0% Δk/k, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 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 ≥ 1.0% Δk/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

INSERT 1

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.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - 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 tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System if the boric acid storage system is OPERABLE, per Specification 3.1.2.6a while in MODE 4, or per Specification 3.1.2.5a while in MODE 5 or 6, 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 is OPERABLE per Specification 3.1.2.6b while in MODE 4, or per Specification 3.1.2.5b while in MODE 5 or 6.

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

INSERT 1

a. When the boric acid tank is a required water source, by verifying at least once per 7 days that:

(1) The flow path from the boric acid tank to the boric acid transfer pump, the boric acid transfer pump, and the recirculation path from the boric acid transfer pump to the boric acid tank is ≥ 63°F, and

INSERT 1

(2) The flow path between the boric acid transfer pump recirculation line to the charging pump suction line is ≥ 50°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.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. A flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two 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 Each of the above required flow paths shall be demonstrated OPERABLE:

INSERT 1

a. By verifying ~~at least once per 7 days~~ that:

(1) The flow path from the boric acid tank to the boric acid transfer pump and from the recirculation line back to the boric acid tank is ≥ 63°F, and

(2) the flow path between the boric acid tank recirculation line to the charging pump suction line is ≥ 50°F,

INSERT 1

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.

SURVEILLANCE REQUIREMENTS (Continued)

INSERT 1

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 safety injection test signal.

INSERT 1

d. At least once per 18 months by verifying that the flow path required by specification 3.1.2.2.a delivers at least 33 gpm to the Reactor Coolant System.

* A one time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.

BORATED WATER SOURCES - 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 system with:

1. A minimum contained volume of 2,600 gallons,

2. Between 6,560 and 6,990 ppm of boron, and,

3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 37,000 gallons,

2. A minimum boron concentration of 2300 ppm, and

3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

a. For the boric acid storage system, when it is the source of borated water at least once per 7 days by:

INSERT 1

1. Verifying the boron concentration of the water,

2. Verifying the water level of the tank, and

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

SURVEILLANCE REQUIREMENTS (Continued)

INSERT 1

b. For the refueling water storage tank by:

1. Verifying the boron concentration at least once per 7 days,

2. Verifying the borated water volume at least once per 7 days, and

3. Verifying the solution temperature at least once per 24 hours, when it is the source of borated water and the outside air temperature is less than 35°F.

[INFORMATION ONLY]

REACTIVITY CONTROL SYSTEMS

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 specifications 3.1.2.1 and 3.1.2.2:

a. A boric acid storage system with:

1. A contained volume of borated water in accordance with figure 3.1.2,

2. A boron concentration in accordance with figure 3.1-2, and

3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A contained volume of between 364,500 and 400,000 gallons of water,

2. A boron concentration of between 2,300 and 2,500 ppm, and

3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required boration water systems, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta K/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. For the boric acid storage system, when it is the source of borated water at least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the water level of each water source, and

3. Verifying the boric acid storage system solution temperature.

b. For the refueling water storage tank by:

1. Verifying the boron concentration at least once per 7 days,

2. Verifying the borated water volume at least once per 7 days, and

3. Verifying the solution temperature at least once per 24 hours when the outside air temperature is less than 35°F.

INSERT 1

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.

c) A core power distribution measurement is obtained and $F_Q(Z)$ $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.c.3.a and 3.1.3.1.c.3.c above are demonstrated.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the limits established in the limiting condition for operation at least once per 12 hours (allowing for one hour thermal soak after rod motion) except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.*

INSERT 1

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

* During Cycle 14, the position of Rod 1SB2 will be determined indirectly by the movable incore detectors within 8 hours following its movement until the repair of the indication system for this rod. During reactor startup, the fully withdrawn position of Rod 1SB2 will be determined by current traces and subsequently verified by the movable incore detectors prior to entry into Mode 1.

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

1. Verify that all analog 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 18 steps when reactor power is ≤ 85% RATED THERMAL POWER or if reactor power is > 85% RATED THERMAL POWER, 12 steps of each other at least once per 8 hours, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.2.1.1 Each analog rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 18 steps when reactor power is ≤ 85% RATED THERMAL POWER or if reactor power is > 85% RATED THERMAL POWER, 12 steps (allowing for one hour thermal soak after rod motion) 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 rod position indication system at least once per 4 hours.

INSERT 1

4.1.3.2.1.2 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of a CHANNEL calibration at least once per 18 months.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 230 steps withdrawn shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

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

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 & 2.

ACTION:

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

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to ≤71% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

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

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

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. ~~At least once per 18 months.~~

INSERT 1

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be FULLY WITHDRAWN.

APPLICABILITY: MODES 1*, and 2*#@

ACTION:

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

a. FULLY WITHDRAW the rod, or,

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

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be FULLY WITHDRAWN by use of the group demand counters, and verified by the analog rod position indicators**,***:

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

b. At least once per 12 hours thereafter.

INSERT 1

* See Special Test Exceptions 3.10.2 and 3.10.3

**For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

*** During Cycle 14, the position of Rod 1SB2 will be determined indirectly by the movable incore detectors within 8 hours following its movement until the repair of the indication system for this rod. During reactor startup, the fully withdrawn position of Rod 1SB2 will be determined by current traces and subsequently verified by the movable incore detectors prior to entry into Mode 1.

With Keff greater than or equal to 1.0

@ Surveillance 4.1.3.4.a is applicable prior to withdrawing control banks in preparation for startup (Mode 2).

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1*, and 2*#

ACTION:

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

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

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

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

SURVEILLANCE REQUIREMENTS

INSERT 1

4.1.3.5 The position of each control bank shall be determined to be within the insertion limits ~~at least once per 12 hours~~ by use of the group demand counters and verified by the analog rod position indicators** 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 3.10.2 and 3.10.3

** For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

\# With K_{eff} greater than or equal to 1.0

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR and ACTION a.2.a)1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the limits specified in the COLR for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE 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:

INSERT 1

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

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

INSERT 1

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

INSERT 1

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside of the target band. Penalty deviation outside of the target band shall be accumulated on a time basis of:

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

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the limits at THERMAL POWER levels below 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

INSERT 1

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.

INSERT 1

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 4.2.1.3 above or by linear interpolation between the most recently measured value and 0 percent at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

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:

1. When THERMAL POWER is ≤ 25%, but > 5% of RATED THERMAL POWER, or

2. When the Power Distribution Monitoring System (PDMS) is inoperable;

and increasing the Measured $F_Q(Z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS or the moveable incore detectors when THERMAL POWER is > 25% of RATED THERMAL POWER, and increasing the measured $F_Q(Z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

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

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

2. The relationship:

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

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the power factor multiplier for F_{xy} in the COLR, 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 core power distribution measurements shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L:

a) Either 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

[INFORMATION ONLY]

INSERT 1

SURVEILLANCE REQUIREMENTS (Continued)

b) At least once per 31 EFPD, whichever occurs first.

2. When the F^C_{xy} is less than or equal to the F^{RTP}_{xy} limit for the appropriate measured core plane, additional core power distribution measurements shall be taken and F^C_{xy} compared to F^{RTP}_{xy} and F^L_{xy} at least once per 31 EFPD.

e. The F_{xy} limit for Rated Thermal Power (F^{RTP}_{xy}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in the COLR per specification 6.9.1.9.

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

1. Lower core region from 0 to 15% inclusive.
2. Upper core region from 85 to 100% inclusive.
3. Grid plane regions at 17.8 ± 2%, 32.1 ±2%, 46.4 ±2%, 60.6 ±2%, and 74.9 ±2% inclusive.
4. Core plane regions within ±2% of core height (±2.88 inches) about the bank demand position of the bank "D" control rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever F^C_{xy} exceeds F^L_{xy}.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by obtaining a core power distribution measurement:

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. INSERT 1

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by the applicable $F^N_{\Delta H}$ uncertainties specified in the COLR.

LIMITING CONDITION FOR OPERATION (Continued)

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 ≤ 55% of RATED THERMAL POWER within the next 4 hours.

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 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. 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 ≤ 55% of RATED THERMAL POWER within the next 4 hours.

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

SURVEILLANCE REQUIREMENTS

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

INSERT 1

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

INSERT 1

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

INSERT 1

c. Obtaining a core power distribution measurement to determine the QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER is > 75 percent of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

INSERT 1

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within the limits of Table 3.2-1 by performing a precision heat balance within 24 hours after achieving steady state conditions ≥90% RATED THERMAL POWER at least once per 18 months. The provisions of Specification 4.0.4 are not applicable.

INSERT 1

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

specified in the Surveillance Frequency Control Program unless otherwise noted

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations ~~during the MODES and~~ at the frequencies ~~shown~~ in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total interlock function shall be demonstrated OPERABLE ~~at least once per 18 months~~ during CHANNEL CALIBRATION testing of ~~each~~ channel affected by interlock operation.

INSERT 1

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME ~~of each reactor trip~~ function shall be verified to be within its limit ~~at least once per 18 months~~. Neutron detectors are exempt from response time testing. ~~Each verification shall include at least one logic train such that both logic trains are verified at least once per 36 months and one channel per function such that all channels are verified 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 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (15)	CHANNEL CALIBRATION (15)	CHANNEL FUNCTIONAL TEST (15)	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip Switch	N.A.	N.A.	~~R~~(9)	1, 2, and *
2. Power Range, Neutron Flux	~~S~~	~~D~~(2), ~~M~~(3) ~~and Q~~(6)	~~Q~~	1, 2, and 3*
3. Power Range, Neutron Flux, High Positive Rate	N.A.	~~R~~(6)	~~Q~~	1, 2
4. Deleted				
5. Intermediate Range, Neutron Flux	~~S~~	~~R~~(6)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	~~S~~(7)	~~R~~(6)	~~Q~~ and S/U(1) (16)	2, 3, 4, 5 and *
7. Overtemperature ΔT	~~S~~	~~R~~	~~Q~~	1, 2
8. Overpower ΔT	~~S~~	~~R~~	~~Q~~	1, 2
9. Pressurizer Pressure--Low	~~S~~	~~R~~	~~Q~~	1, 2
10. Pressurizer Pressure--High	~~S~~	~~R~~	~~Q~~	1, 2
11. Pressurizer Water Level--High	~~S~~	~~R~~	~~Q~~	1, 2
12. Loss of Flow - Single Loop	~~S~~	~~R~~	~~Q~~	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (15)	CHANNEL CALIBRATION (15)	CHANNEL FUNCTIONAL TEST (15)	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow Two Loops	~~S~~	~~R~~	N.A.	1
14. Steam Generator Water Level--Low-Low	~~S~~	~~R~~	~~Q~~	1, 2
15. DELETED				
16. Undervoltage - Reactor Coolant Pumps	N.A.	~~R~~	~~Q~~	1
17. Underfrequency - Reactor Coolant Pumps	N.A.	~~R~~	~~Q~~	1
18. Turbine Trip				
a. Low Autostop Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
b. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	~~M~~(4)(5)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	~~R~~	1
21. Reactor Trip Breaker	N.A.	N.A.	~~M~~(5)(11)(13) ~~and R~~(14)	1, 2 and *
22. Automatic Trip Logic	N.A.	N.A.	~~M~~(5)	1, 2 and *

TABLE 4.3-1 (Continued)

NOTATION

* With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference ≥ 3 percent. INSERT 4

(4) - Manual SSPS functional input check ~~every 18 months~~. **

(5) - Each train or logic channel shall be tested ~~at least every 62 days on a STAGGERED TEST BASIS~~. INSERT 1

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

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - Deleted

(9) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip mechanism for the Manual Reactor Trip Function.

The Test shall also verify OPERABILITY of the Bypass Breaker Trip circuits.

(10) - DELETED

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Reactor Trip Breaker Undervoltage and Shunt Trip mechanisms.

(12) - DELETED

** A one time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.

TABLE 4.3-1 (Continued)

NOTATION

(13) - Verify operation of Bypass Breakers Shunt Trip function from local pushbutton while breaker is in the test position prior to placing breaker in service.

(14) - Perform a functional test of the Bypass Breakers U.V. Attachment via the SSPS.

(15) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

(16) At the frequency specified in the Surveillance Frequency Control Program

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

Specified in the Surveillance Frequency Control Program unless otherwise noted

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

INSERT 1

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit at least once per 18 months. Each verification shall include at least one logic train such that both logic trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3. The provisions of Specification 4.0.4 are not applicable to MSIV closure time testing. The provisions of Specification 4.0.4 are not applicable to the turbine driven auxiliary feedwater pump provided the surveillance is performed within 24 hours after the secondary steam generator pressure is greater than 680 psig.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (7)	CHANNEL CALIBRATION (7)	CHANNEL FUNCTIONAL TEST (7)	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION				
a. Manual Initiation	N.A.	N.A.	~~R*~~	1,2,3,4
b. Automatic Actuation Logic	N.A.	N.A.	~~M~~(2)	1,2,3,4
c. Containment Pressure-High	~~S~~	~~R~~	~~Q~~(3)	1,2,3
d. Pressurizer Pressure--Low	~~S~~	~~R~~	~~Q~~	1,2,3
e. Differential Pressure Between Steam Lines--High	~~S~~	~~R~~	~~Q~~	1,2,3
f. Steam Flow in Two Steam Lines--High coincident with Tavg--Low-Low or Steam Line Pressure-Low	~~S~~	~~R~~	~~Q~~	1,2,3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	~~R~~	1,2,3,4
b. Automatic Actuation Logic	N.A.	N.A.	~~M~~(2)	1,2,3,4
c. Containment Pressure--High-High	~~S~~	~~R~~	~~Q~~(3)	1,2,3

* A one time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (7)	CHANNEL CALIBRATION (7)	CHANNEL FUNCTIONAL TEST (7)	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONTAINMENT ISOLATION				
a. Phase "A" Isolation				
1. Manual	N.A.	N.A.	~~R~~	1,2,3,4
2. From Safety Injection Automatic Actuation Logic	N.A.	N.A.	~~M~~(2)	1,2,3,4
b. Phase "B" Isolation				
1. Manual	N.A.	N.A.	~~R~~	1,2,3,4
2. Automatic Actuation Logic	N.A.	N.A.	~~M~~(2)	1,2,3,4
3. Containment Pressure--High-High	~~S~~	~~R~~	~~Q~~(3)	1,2,3
c. Containment Ventilation Isolation				
1. Manual	N.A.	N.A.	~~R~~	1,2,3,4
2. Automatic Actuation Logic	N.A.	N.A.	~~M~~(2)	1,2,3,4
3. Containment Atmosphere Gaseous Radioactivity - High		Per ~~table 4.3-3~~ Surveillance Requirement 4.3.3.1		

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (7)	CHANNEL CALIBRATION (7)	CHANNEL FUNCTIONAL TEST (7)	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	R	1,2,3**
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1,2,3
c. Containment Pressure--High-High	S	R	Q(3)	1,2,3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low or Steam Line Pressure--Low	S	R	Q	1,2,3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	Q	1,2,3
6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC) LOGIC				
a. Inputs	N.A.	N.A.	M(6)	1,2,3,4
b. Logic, Timing and Outputs *	N.A.	N.A.	M(1)	1,2,3,4
7. UNDERVOLTAGE, VITAL BUS				
a. Loss of Voltage	S	R	M	1,2,3
b. Sustained Degraded Voltage	S	R	M	1,2,3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (7)	CHANNEL CALIBRATION (7)	CHANNEL FUNCTIONAL TEST (7)	MODES IN WHICH SURVEILLANCE REQUIRED
8. AUXILIARY FEEDWATER				
a. Automatic Actuation Logic	N.A.	N.A.	~~M~~(2)	1,2,3
b. NOT USED				
c. Steam Generator Water Level--Low-Low	~~S~~	~~R~~	~~Q~~	1,2,3
d. Undervoltage - RCP	~~S~~	~~R~~	~~Q~~	1,2
e. S.I.	See 1 above (All S.I. surveillance requirements)			
f. Trip of Main Feedwater Pumps	N.A.	N.A.	~~R~~	1
g. Station Blackout	See 6b and 7 above (SEC and U/V Vital Bus)			

TABLE 4.3-2 (Continued)

TABLE NOTATION

* Outputs are up to, but not including, the output relays.

** The provisions of Specification 4.0.4 are not applicable.

INSERT 1

(1) Each logic channel shall be tested at least once per 62 days on a STAGGERED TEST BASIS. The CHANNEL FUNCTION TEST of each logic channel shall verify that its associated diesel generator automatic load sequence timer is OPERABLE with the interval between each load block within 1 second of its design interval.

(2) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

(4) NOT USED

(5) NOT USED

(6) Inputs from Undervoltage, Vital Bus, shall be tested monthly. Inputs from Solid State Protection System shall be tested every 62 days on a STAGGERED TEST BASIS.

(7) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

INSERT 1

Table 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNELS CHECKS	SOURCE CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS					
a. Fuel Storage Area	S	M	R	Q	*
2. PROCESS MONITORS					
a. Containment Monitors					
1) Gaseous Activity					
a) Purge & Pressure Vacuum Relief Isolation	S	M	R	Q	1, 2, 3, 4 & 5
b) RCS Leakage Detection	S	M	R	Q	1, 2, 3 & 4
2) Air Particulate Activity					
a) (NOT USED)					
b) RCS Leakage Detection	S	M	R	Q	1, 2, 3 & 4

*With fuel in the storage pool or building.

DELETED

(Delete this page)

TABLE 4.3-3 (Continued)
RADIATION MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNELS CHECKS	SOURCE CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	S	M	R	Q	1, 2, 3 & 4
2) High Range Auxiliary Building Exhaust System (Plant Vent)	S	M	R	Q	1, 2, 3 & 4
3) Condenser Exh. Sys.	S	M	R	Q	1, 2, 3 & 4
3. CONTROL ROOM					
a. Air Intake - Radiation Level	S	M	R	Q	**

** ALL MODES and during movement of irradiated fuel assemblies and during CORE ALTERATIONS.

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, either restore the inoperable channel to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

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

INSERT 1

DELETED

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Pressurizer Pressure	M	R
2. Pressurizer Level	M	R
3. Steam Generator Pressure	M	R
4. Steam Generator Level	M	R

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be operable.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. As shown in Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-11.

specified in the Surveillance Frequency Control Program unless otherwise noted

TABLE 4.3-11
SURVEILLANCE REQUIREMENTS FOR ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECKS (1)	CHANNEL CALIBRATION (1)	CHANNEL FUNCTIONAL TEST (1)
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	~~M~~	~~R~~	N.A.
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	~~M~~	~~R~~	N.A.
3. Reactor Coolant Pressure (Wide Range)	~~M~~	~~R~~	N.A.
4. Pressurizer Water Level	~~M~~	~~R~~	N.A.
5. Steam Line Pressure	~~M~~	~~R~~	N.A.
6. Steam Generator Water Level (Narrow Range)	~~M~~	~~R~~	N.A.
7. Steam Generator Water Level (Wide Range)	~~M~~	~~R~~	N.A.
8. Refueling Water Storage Tank Water Level	~~M~~	~~R~~	N.A.
9. deleted			
10. Auxiliary Feedwater Flow Rate	S/U#	~~R~~	N.A.
11. Reactor Coolant System Subcooling Margin Monitor	~~M~~	N.A.* INSERT 1	N.A.

#Auxiliary Feedwater System is used on each startup and flow rate indication is verified at that time.

*The instruments used to develop RCS subcooling margin are calibrated ~~on an 18 month cycle~~; the monitor will be compared ~~quarterly~~ with calculated subcooling margin for known input values.

INSERT 2

TABLE 4.3-11 (Continued)
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK (1)	CHANNEL CALIBRATION (1)	CHANNEL FUNCTIONAL TEST (1)
12. PORV Position Indicator	M	N.A.	R
13. PORV Block Valve Position Indicator	M	N.A.	Q*
14. Pressurizer Safety Valve Position Indicator	M	N.A.	R
15. Containment Pressure - Narrow Range	M	R	N.A.
16. Containment Pressure - Wide Range	M	R	N.A.
17. Containment Water Level - Wide Range	M	R**	N.A.
18. Core Exit Thermocouples	M	R	N.A.
19. Reactor Vessel Level Instrumentation System (RVLIS)	M	R	N.A.
20. Containment High Range Accident Radiation Monitor	S	R	Q
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor	S	R	Q

*Unless the block valve is closed in order to meet the requirements of Action b, or c in specification 3.4.3.

Table Notation

(1) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

** A one-time extension to this surveillance requirement is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be operable to ensure that the limits of ODCM Control 3.11.1.1 are not exceeded.

APPLICABILITY: At all times.

ACTION:

a. Not Used

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Exert best efforts to return the instrument to OPERABLE status within 30 days and, if unsuccessful, explain in the next annual radioactive effluent release report why the inoperability was not corrected in a timely manner.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies ~~shown~~ in Table 4.3-12.

specified in the Surveillance Frequency Control Program unless otherwise noted

TABLE 4.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK (1)	SOURCE CHECK (1)	CHANNEL CALIBRATION (1)	CHANNEL FUNCTIONAL TEST (1)
1. Not Used				
2. Not Used				
3. Not Used				
4. TANK LEVEL INDICATING DEVICES**				
a. Temporary Outside Storage Tanks as Required	D*	N.A.	~~R~~	~~Q~~

TABLE NOTATION

* During liquid additions to the tank.

** If tank level indication is not provided, verification will be done by visual inspection.

(1) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILTY: MODE 1, above 25% RATED THERMAL POWER (RTP)

ACTION:

With any of the operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, or 3.3.3.14.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system, satisfying the OPERABILITY requirements listed in Specification 3.3.3.2, to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.14.1 The operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, and 3.3.3.14.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

4.3.3.14.2 Calibration of the PDMS is required: INSERT 1

a. At least once every 180 Effective Full Power Days when the minimum number and core coverage criteria as defined in 3.3.3.14.b.1 and 3.3.3.14.b.2 are satisfied, or

INSERT 1

b. At least once every 31 Effective Full Power Days when only the minimum number criterion as defined in 3.3.3.14.b.3 is satisfied.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS

NORMAL 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 1 hour.

SURVEILLANCE REQUIREMENT

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

INSERT 1

*See Special Test Exception 3.10.4.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

INSERT 1

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 5% (narrow range) at least once per 12 hours.

INSERT 1

*All reactor coolant pumps may be de-energized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration (2) core outlet temperature is maintained at least 10°F below saturation temperature, and (3) the rod control system is de-energized**

**The rod control system shall be considered de-energized when one or more of the following conditions exist:

1) Both Rod Drive MG set motor breakers are open.

2) Both Rod Drive MG set generator breakers are open.

3) A combination of at least three of the Reactor Trip and/or Reactor Trip Bypass Breakers are open.

If none of the above conditions for de-energizing the rod control system are met; the system shall be considered energized.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per the inservice testing schedule.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to 5% of narrow range at least once per 12 hours.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

INSERT 1

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.4 Two# residual heat removal loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5.##

ACTION:

a. With less than the above required loops operable, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no RHR 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 RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

INSERT 1

One RHR loop may be inoperable for up to two hours for surveillance testing, provided the other RHR loop is OPERABLE and in operation. Additionally, four filled reactor coolant loops, with at least two steam generators with their secondary side water levels greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop.

A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 312 °F unless 1) the pressurizer water volume is less than 1650 cubic feet (equivalent to approximately 93.2% of level), or 2) the secondary water temperature of each steam generator is less than 50 °F above each of the RCS cold leg temperatures.

* Systems supporting RHR loop operability may be excepted as follows:

a. The normal or emergency power source may be inoperable.

** The residual heat removal pumps may be de-energized for up to 2 hours 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.

3/4.4.3 RELIEF VALVES

INSERT 1

SURVEILLANCE REQUIREMENTS

4.4.3.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

b. Operating solenoid valves, air control valves, and check valves on associated air accumulators in PORV control systems through one complete cycle of full travel, and

c. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.3.2 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 b, or c in Specification 3.4.3.

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1650 cubic feet (92% indicated level), and at least two groups of pressurizer heaters each having a capacity of ≥ 150 kw and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.4.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring circuit current at least once each refueling outage.

INSERT 1

4.4.4.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.

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 particulate radioactivity monitoring system,

b. The containment sump level monitoring system, and

c. Either the containment fan cooler condensate flow rate or the containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) monitoring systems-performance of CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

INSERT 1

b. Containment sump level and containment fan cooler condensate flow rate (if being used) monitoring systems-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. 150 gallons per day primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE or primary-to-secondary leakage not within limit, 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 primary-to-secondary leakage, 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.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by;

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory at least once per 12 hours.

INSERT 1

SURVEILLANCE REQUIREMENTS (Continued)

INSERT 1

c.* Verifying primary-to-secondary leakage is ≤ 150 gallons per day through any one steam generator at least once per 72 hours during steady state operation.

d.* Performance of a Reactor Coolant System water inventory balance** at least once per 72 hours. The water inventory balance shall be performed with the plant at steady state conditions. The provisions of specification 4.0.4 are not applicable for entry into Mode 4, and

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

*Not required to be completed until 12 hours after establishment of steady state operation.

**Not applicable to primary-to-secondary leakage.

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES LIMITING CONDITION FOR OPERATION

3.4.6.3 Reactor Coolant System Pressure Isolation Valves specified in Table 4.4-3 shall be OPERABLE.

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

ACTION:

a. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the specified limit in Table 4.4-3, 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.

SURVEILLANCE REQUIREMENTS

4.4.6.3 Each Reactor Coolant System Pressure Isolation Valve specified in Table 4.4-3 shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying ~~leakage to be within~~ its limit:

a. At least once per 18 months. INSERT 1

b. 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.

c. Prior to returning the valve to service following maintenance repair or replacement work on the valve.

d. For the Residual Heat Removal and Safety Injection Systems hot and cold leg injection valves and accumulator valves listed in Table 4.4-3 the testing will be done within 24 hours following valve actuation due to automatic or manual action or flow through the valve. For all other systems testing will be done once per refueling.

The provisions of specification 4.0.4 are not applicable for entry into MODE 3 or 4.

[INFORMATION ONLY]

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. ≤ 1.0 µCi/gram DOSE EQUIVALENT I-131, and

b. ≤ 100/ÊµCi/gram.

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

ACTION:

MODES 1, 2 and 3*

a. With the specific activity of the primary coolant > 1.0 µCi/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} < 500°F within 6 hours.

b. With the specific activity of the primary coolant > 100/ÊµCi/gram, be in at least HOT STANDBY with T_{avg} < 500°F within 6 hours.

c. LCO 3.0.4.c is applicable.

MODES 1, 2, 3, 4 and 5

a. With the specific activity of the primary coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 or > 100/ÊµCi/gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary 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} ≥ 500°F.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

INSERT 1

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ 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 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 & 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

#Until the specific activity of the primary coolant system is restored within its limits.
*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.4.9.1.1 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.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

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

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties 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.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each POPS shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the POPS actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the POPS is required OPERABLE, and at least once per 31 days thereafter when the POPS is required OPERABLE.

INSERT 1

b. Performance of a CHANNEL CALIBRATION on the POPS actuation channel at least once per 18 months.

c. Verifying the POPS isolation valve is open at least once per 72 hours when the POPS is being used for overpressure protection.

d. Testing pursuant to Specification 4.0.5.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vents(s) is being used for overpressure protection.

INSERT 1

*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.

3/4.4.12 HEAD VENTS

LIMITING CONDITION FOR OPERATION

3.4.12 Four reactor vessel head vent paths shall be operable with the vent paths closed. A vent path consists of at least two head vent valves in series, powered from vital sources, and associated flowpath.

APPLICABILITY: MODES 1, 2, 3 AND 4.

ACTION:

a. With one, two or three reactor vessel head vent path(s) inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path(s) is maintained closed with the valve actuators key locked in the closed position; restore the inoperable vent path(s) 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 four reactor vessel head vent paths inoperable; maintain the inoperable vent paths 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

4.4.12 Reactor vessel head vent system vent paths shall be demonstrated OPERABLE ~~at least once per 18 months~~ by:

INSERT 1

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each valve in the vent paths through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

3. Verifying flow through the reactor vessel head vent system vent path during venting.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained volume of between 6,223 and 6,500 gallons of borated water,

c. A boron concentration of between 2,200 and 2,500 ppm, and,

d. A nitrogen cover-pressure of between 595.5 and 647.5 psig.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve or boron concentration outside the required limits, restore the inoperable accumulator to OPERABLE status within 24 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within 24 hours and be in HOT SHUTDOWN within the next 12 hours.

c. With the boron concentration of one accumulator outside the required limits, restore the boron concentration to within the required limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than or equal to 1000 psig within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. ~~At least once per 12 hours~~ by:

INSERT 1

1. Verifying the water level and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

---

* Pressurizer Pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued) INSERT 1

b. At least once per 31 days and within 6 hours after each solution volume increase of ≥ 1% of tank volume by verifying the boron concentration of the accumulator solution.

c. At least once per 31 days when the RCS pressure is greater than 1000 psig by verifying that the power lockout switch is in lockout.

d.* At least once per 18 months by verifying that each accumulator isolation valve opens automatically upon receipt of a safety injection test signal.

* A one time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. 1 SJ 69	a. RHR pump suction	a. open
b. 1 SJ 30	b. SI pump suction	b. open
c. 11 SJ 40	c. SI discharge to hot legs	c. closed
d. 12 SJ 40	d. SI discharge to hot legs	d. closed
e. 1 RH 26	e. RHR discharge to hot legs	e. closed
f. 11 SJ 49	f. RHR discharge to cold legs	f. open
g. 12 SJ 49	g. RHR discharge to cold legs	g. open
h. 1 CS 14#	h. Spray additive tank discharge	h. open
i. 1 SJ 135	i. SI discharge to cold legs	i. open
j. 1 SJ 67	j. SI recirc. line isolation	j. open
k. 1 SJ 68	k. SI recirc. line isolation	k. open
l. 11 SJ 44	l. Containment sump isolation valve	l. closed
m. 12 SJ 44	m. Containment sump isolation valve	m. closed

INSERT 1

2. Verifying that the following valves are in the indicated positions:

Valve Number	Valve Function	Valve Position
a. 11 RH 19	a. RHR crosstie valve	a. open
b. 12 RH 19	b. RHR crosstie valve	b. open

b. At least once per 31 days by:

1. 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.

2. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points.

If inoperable, the applicable Technical Specification is 3.6.2.2.

SURVEILLANCE REQUIREMENTS (Continued)

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. At least once daily (24 hour consecutive period) the areas affected within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

INSERT 1

d. At least once per 18 months by:

1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, 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 each of the following pumps start automatically upon receipt of a safety injection test signal:

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

* A one time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.

SURVEILLANCE REQUIREMENTS (Continued)

f. By verifying that each of the following pumps develops the indicated Total Dynamic Head (TDH) when tested at the test flow point pursuant to Specification 4.0.5:

1. Centrifugal charging pump ≥ 2338 psi TDH

2. Safety Injection Pump ≥ 1369 psi TDH

3. Residual heat removal pump ≥ 165 psi TDH

g. By verifying the correct position of each of 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.

2. ~~At least once per 18 months~~

INSERT 1

HPSI SYSTEM VALVE NUMBER	LPSI SYSTEM VALVE NUMBER
11 SJ 16	11 SJ 138
12 SJ 16	12 SJ 138
13 SJ 16	13 SJ 138
14 SJ 16	14 SJ 138
	11 SJ 143
	12 SJ 143
	13 SJ 143
	14 SJ 143

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 Safety Injection pumps, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is ≥ 453 gpm; and

b) The total flow rate through all four injection lines is ≤ 647 gpm, and

c) The difference between any pair of injection line flow rates is ≤ 12.0 gpm, and

d) The total pump flow rate is ≤ 664 gpm in the cold leg alignment, and

e) The total pump flow rate is ≤ 654 gpm in the hot leg alignment.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All safety injection pumps and centrifugal charging pumps, except the above required OPERABLE pump, shall be demonstrated to be inoperable at least once per 12 hours while in MODE 4 and the temperature of one or more of the RCS cold legs is less than or equal to 312°F, MODE 5, or MODE 6 when the head is on the reactor vessel by either of the following methods:

a. By verifying that the motor circuit breakers have been removed from their electrical power supply circuits or,

INSERT 1

b. For verifying that the pump is in a recirculation flow path and that two independent means of preventing RCS injection are utilized.

SEAL INJECTION FLOW

LIMITING CONDITION FOR OPERATION

3.5.4 Reactor coolant pump seal injection flow shall be ≤40 gpm with centrifugal charging pump discharge header pressure ≥2430 psig and the charging flow control valve full open.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With seal injection flow not within the limit, adjust manual seal injection throttle valves to give a flow within the limit with the charging pump discharge pressure ≥2430 psig and the charging flow control valve full open within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.5.4 ~~At least once per 31 days~~, verify manual seal injection throttle valves are adjusted to give a flow within the limit with centrifugal charging pump discharge header pressure ≥2430 psig, and the charging flow control valve full open.

The provisions of Specification 4.0.4 are not applicable for entry into Mode 3. This exemption is allowed for up to 4 hours after the Reactor Coolant System pressure stabilizes at 2235 ± 20 psig.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A contained volume of between 364,500 and 400,000 gallons of borated water.

b. A boron concentration of between 2300 and 2500 ppm, and

c. A minimum water temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, 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.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

INSERT 1

1. Verifying the water level 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 < 35°F.

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 one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a1. At least once per 31 days by verifying that each containment manual valve or blind flange that is located outside containment and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls. Valves and blind flanges in high radiation areas may be verified by use of administrative controls.

a2. Prior to entering Mode 4 from Mode 5 if not performed within the last 92 days by verifying that each containment manual valve or blind flange that is located inside containment and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls. Valves and blind flanges in high radiation areas may be verified by use of administrative controls.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

INSERT 1

c. At least once per 12 hours by verifying that the surveillance requirements of 4.6.2.3.a are met for penetrations associated with the containment fan coil units.

d. At least once per 18 months by verifying that the surveillance requirements of 4.6.2.3.d are met for penetrations associated with the containment fan coil units.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATIONS (Continued)

c. One or more containment air locks inoperable for reasons other than condition a. or b.

   1. Immediately initiate action to evaluate overall containment leakage per LCO 3.6.1, and:

   2. Verify that at least one door is closed in the affected air lock within 1 hour, and:

   3. Restore the air lock to OPERABLE status within 24 hours.

d. If the ACTIONS and associated completion times of a., b., or c. cannot be met, be in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying seal leakage rate in accordance with the Containment Leakage Rate Testing program.

b. By conducting an overall air lock leakage test in accordance with the Containment Leakage Rate Testing Program.

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

INSERT 1

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -1.5 and +0.3 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 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 shall be determined to be within the limits ~~at least once per 12 hours.~~

INSERT 1

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature > 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 Verify the Containment Average Air Temperature is within limit at least once per twenty four hours.

INSERT 1

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 RHR pump discharge.

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 204 psid when tested pursuant to Specification 4.0.5.

INSERT 1

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 High-High pressure test signal.

2. Verifying that each spray pump starts automatically on a Containment High-High pressure test signal.

d. Following activities that could result in nozzle blockage, either evaluate the work performed to determine the impact to the containment spray system, or perform 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 2568 and 4000 gallons of between 30 and 32 percent by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the spray additive system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

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

b. It least once per 6 months by:

1. Verifying the solution level in the tank, and

2. Verifying the concentration of the NaOH solution by chemical analysis.

INSERT 1

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 High-High pressure test signal.

d. At least once per 5 years by:

1. Verifying a NaOH solution flow rate of 12 ± 3 gpm from the spray additive tank through sample valve 1CS61 with the spray additive tank at 2.5 ± 0.5 psig and

[INFORMATION ONLY]

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the spray additive tank eductor flow will be 35 ± 3.5 gpm to each containment spray system. Testing may be performed by measuring the flow of borated water from the RWST through the installed 2" test line and Valve CS31; using this test line up with the spray pump operating in the recirculation mode and the RWST level at 41 feet ± 0.5 feet, the measured flow shall be 57 gpm ± 5.7 gpm.

LIMITING CONDITION FOR OPERATION

3.6.2.3 Five containment cooling fans shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or two of the above required containment cooling fans inoperable, restore the inoperable cooling fan(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With three or more of the above required containment cooling fans inoperable, restore at least three cooling fans 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. Restore the remaining inoperable cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment cooling fan shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 12 hours by:

1. Verifying the water level in each service water accumulator vessel is greater than or equal to 226 inches and less than or equal to 252 inches.

2. Verifying the temperature in each service water accumulator vessel is greater than or equal to 55°F and less than or equal to 95°F.

3. Verifying the nitrogen cover pressure in each service water accumulator vessel is greater than or equal to 135 psig and less than or equal to 160 psig.

b. At least once per 31 days by:

INSERT 1

1. Starting (unless already operating) each fan from the control room in low speed.

2. Verifying that each fan operates for at least 15 minutes in low speed.

3. Verifying a cooling water flow rate of greater than or equal to 1300 gpm to each cooler.

c. At least once per 18 months by verifying that on a safety injection test signal:

1. Each fan starts automatically in low speed.

2. The automatic valves and dampers actuate to their correct positions and that the cooling water flow rate to each cooler is greater than or equal to 1300 gpm.

d. At least once per 18 months by verifying that on a loss of offsite power test signal, each service water accumulator vessel discharge valve response time is within limits.

SURVEILLANCE REQUIREMENTS (Continued)

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

INSERT 1

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

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Not used.

d. Verifying that on a Containment Purge and Pressure-Vacuum Relief isolation test signal, each required Purge and each Pressure-Vacuum Relief valve actuates to its isolation position.

INSERT 1

e. Verifying that the Containment Pressure-Vacuum Relief Isolation valves are limited to ≤ 60% opening angle.

4.6.3.1.3 At least once per 18 months, verify that on a main steam isolation test signal, each main steam isolation valve actuates to its isolation position.

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

4.6.3.1.5 Each required containment purge isolation valve shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cyclings, then at least once per 72 hours by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Specification 4.6.1.2.b for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60La.

INSERT 1

4.6.3.1.6 A pressure drop test to identify excessive degradation of resilient valve seals shall be conducted on the:

a. Required Containment Purge Supply and Exhaust Isolation Valves at least once per 6 months.

b. Deleted.

INSERT 1

4.6.3.1.7 The required containment purge supply and exhaust isolation valves shall be determined closed at least once per 31 days.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated manual activation switches in the control room and flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate vital busses, and

b. One 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.

d. LCO 3.0.4.b is not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

INSERT 1

1. 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.

2. Verify the manual maintenance valves in the flow path to each steam generator are locked open.

SURVEILLANCE REQUIREMENTS (continued)

b. ~~At least once per 92 days on a STAGGERED TEST BASIS~~ by:

1. Verify that the developed head of each motor driven pump at the flow test point is greater than or equal to the required developed head.

2. Verify that the developed head of the steam driven pump at the flow test point is greater than or equal to the required developed head when the steam generator pressure is >680 psig. The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 24 hours after secondary side pressure is greater than 680 psig.

INSERT 1

c. ~~At least once per 18 months~~ by:

1. Verifying that each auxiliary feedwater automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

2. Verifying that each auxiliary feedwater pump starts automatically on an actual or simulated actuation signal.

The provisions of Specification 4.0.4 are not applicable to the turbine driven auxiliary feedwater pump provided the surveillance is performed within 24 hours after the secondary steam generator pressure is greater than 680 psig.

AUXILIARY FEED STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The auxiliary feed storage tank (AFST) shall be OPERABLE with a minimum contained volume of 200,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the auxiliary feed storage tank inoperable, within 4 hours either:

a. Restore the AFST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of a demineralized water or a fire protection/domestic water storage tank as a backup supply to the auxiliary feedwater pumps and restore the auxiliary feed storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The auxiliary feed storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the water level is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

INSERT 1

4.7.1.3.2 A demineralized water storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the tank contains ≥ 200,000 gallons of water and by verifying proper alignment of valves for taking suction from this tank when it is the supply source for the auxiliary feedwater pumps.

4.7.1.3.3 A fire protection/domestic water storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the tank contains ≥ 200,000 gallons of water and by verifying proper alignment of valves for taking suction from this tank when it is the supply source for the auxiliary feedwater pumps.

[INFORMATION ONLY].

PLANT SYSTEMS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be ≤0.10 μCi/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the secondary coolant system > 0.10 μCi/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 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

INSERT 1

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.
	b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 70°F when the pressure of either coolant in the steam generator is > 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 ≤ 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generator shall be determined to be < 200 psig at least once per hour when the temperature of either the primary or secondary coolant is < 70°F.

INSERT 1

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

INSERT 1

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two independent service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours * or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two service water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

INSERT 1

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on Safeguards Initiation signal.

* Operation with only the 11 service water loop OPERABLE may continue for up to 10 days. This note is applicable for one time use during Salem Unit No.1 Cycle 15.

PLANT SYSTEMS

3/4.7.5 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.5.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Delaware River exceeds 10.5' Mean Sea Level USGS datum, at the service water intake structure.

APPLICABILITY: At all times.

ACTION:

a. With the water level at the service water intake structure above elevation 10.5' Mean Sea Level USGS datum, close all watertight doors within 2 hours.

b. With the water level at the service water intake structure above elevation 11.5' Mean Sea Level USGS datum, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The water level at the service water intake structure shall be determined to be within the limits by:

INSERT 1

a. Measurement ~~at least once per 24 hours~~ when the water level is below elevation 10.5' Mean Sea Level USGS datum, and

b. Measurement ~~at least once per 2 hours~~ when the water level is equal to or above elevation 10.5' Mean Sea Level USGS datum.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Each control room emergency air conditioning system filtration train shall be demonstrated OPERABLE:

INSERT 1

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train(s) and verifying that the train(s) operates with each fan operating for at least 15 minutes.

b. At least once per 18 months or prior to return to service (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 charcoal adsorbers remove ≥99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the ventilation system at a flow rate of 8000 cfm ± 10%.

2. Verifying that the HEPA filter banks remove ≥99% of the DOP when they are tested in-place while operating the ventilation system at a flow rate of 8000 cfm ± 10%.

3. Verifying within 31 days after removal from the CREACS unit, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the CREACS unit, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

d. At least once per 18 months by:

INSERT 1

1. Verifying that the pressure drop across the combined HEPA filter and charcoal adsorber bank is ≤ 3.5 inches water gauge while operating the ventilation system at a flow rate of 8000 cfm ± 10%.

[INFORMATION ONLY]

SURVEILLANCE REQUIREMENTS (Continued)

2.* Verifying that on a safety injection test signal or control room intake high radiation test signal, the system automatically actuates in the pressurization mode by opening the outside air supply and diverting air flow through the HEPA filter and charcoal adsorber bank.

3. Deleted

4. Verifying that on a manual actuation signal, the system will actuate to the required pressurization or recirculation operating mode.

5. Verify each CREACS train has the capability to remove the assumed heat load.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place while operating the filter system at a flow rate of 8000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal absorber bank by verifying that the charcoal absorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the filter system at a flow rate of 8000 cfm ± 10%.

4.7.6.2 Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program (Refer to T.S. 6.18).

* A one time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.

4.7.7.1 The above required Auxiliary Building Ventilation System shall be demonstrated OPERABLE by:

a. At least once per 12 hours by verifying negative pressure in the Auxiliary Building.

INSERT 1

b. At least once per 31 days by starting each fan, from the control room, and verifying that each fan operates for at least 15 minutes.

c. At least once per 18 months by verifying that the System starts following a Safety Injection Test Signal.

3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.8.1 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 ≥ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limits shall be immediately withdrawn from use and:

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.8.1.2 Test Frequencies - Each category of sealed sources shall be tested at the frequency described below.

a. Sources in use (excluding startup sources and fission detectors previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive materials.

INSERT 1

[INFORMATION ONLY]

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

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 six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

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 and following repair or maintenance to the source or detector.

4.7.8.1.3 Reports - 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 $\geq$ 0.005 microcuries of removable contamination.

LIMITING CONDITION FOR OPERATION

ACTION: MODES 5 and 6 or during movement of irradiated fuel assemblies. *

a. With one chiller inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Restore the chiller to OPERABLE status within 14 days or;

3. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

b. With two chillers inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Align the control room emergency air conditioning system (CREACs) for single filtration operation using the Salem Unit 2 train within 4 hours and;

3. Restore at least one chiller to OPERABLE status within 72 hours or;

4. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

c. With one chilled water pump inoperable, restore the chilled water pump to OPERABLE status within 7 days or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.10 The chilled water loop which services the safety-related loads in the Auxiliary Building shall be demonstrated OPERABLE:

INSERT 1

a. At least once per 31 days by verifying that each manual valve in the chilled water system flow path servicing safety related components 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 actuates to its correct position on a Safeguards Initiation signal.

c. At least once per 92 days by verifying that each chiller starts and runs.

* During Modes 5 and 6 and during movement of irradiated fuel assemblies, chilled water components are not considered to be inoperable solely on the basis that the backup emergency power source, diesel generator, is inoperable.

**A one time extension to this surveillance requirement for performance of relay time response and sequence testing of the safeguard equipment control (SEC) system, which partially satisfies the surveillance requirement, is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.

3/4.7.11 FUEL STORAGE POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.11 The fuel storage pool boron concentration shall be ≥ 800 ppm.

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool.

ACTION:

With fuel storage pool boron concentration not within limit:

a. Immediately suspend movement of fuel assemblies in the fuel storage pool and

b. Initiate action to:

1. immediately restore fuel storage pool boron concentration to within limit or

2. immediately perform a fuel storage pool verification.

c. LCO 3.0.3 is not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11 Verify the fuel storage pool boron concentration is within limit ~~every 7 days~~.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Two physically independent A.C. circuits between the offsite transmission network and the onsite Class 1E distribution system (vital bus system) shall be:

INSERT 1

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, power availability, and

INSERT 1

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) vital bus supply from one 13/4 kv transformer to the other 13/4 kv transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

INSERT 1

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in its day tank.

2. Verifying the diesel generator starts from standby conditions* and achieves ≥ 3910 volts and ≥ 58.8 Hz in ≤ 13 seconds, and subsequently achieves steady state voltage of ≥ 3910 and ≤ 4400 volts and frequency of 60 ± 1.2 Hz.

   Subsequently, verifying the generator is synchronized with voltage maintained ≥ 3910 and ≤ 4580 volts, gradually loaded to 2340-2600 kw**, and operates at a load of 2340-2600 kw for greater than or equal to 60 minutes.

3. Verifying the diesel generator is aligned to provide standby power to the associated vital bus.

INSERT 1

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 one hour by checking for and removing accumulated water from the day tanks.

INSERT 1

c. At least once per 6 months by verifying the diesel generator starts from standby conditions* and achieves ≥ 3910 volts and ≥ 58.8 Hz in ≤ 13 seconds, and subsequently achieves steady state voltage of ≥ 3910 and ≤ 4400 volts and frequency of 60 ± 1.2 Hz.

   The generator shall be synchronized to its emergency bus with voltage maintained ≥ 3910 and ≤ 4580 volts, loaded to 2340-2600** kw in less than or equal to 60 seconds, and operate at a load of 2340-2600 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.2, may also serve to concurrently meet those requirements.

INSERT I

SURVEILLANCE REQUIREMENTS (Continued)

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

1. DELETED

2. Verifying that, on rejection of a load greater than or equal to 820 kw, the voltage and frequency are restored to ≥ 3910 and ≤ 4400 volts and 60 ± 1.2 Hz within 4 seconds, and subsequently achieves a steady state frequency of ≥ 58.8 and ≤ 60.5 Hz.

3. Simulating a loss of offsite power by itself, and:

   a) Verifying de-energization of the vital bus and load shedding from the vital bus.

   b) Verifying the diesel starts on the auto-start signal*, energizes the vital bus with permanently connected loads within 13 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 of these loads, the steady state voltage and frequency of the vital bus shall be maintained at ≥ 3910 and ≤ 4400 volts and ≥ 58.8 and ≤ 60.5 Hz during this test.

4. 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 diesel generator shall achieve ≥ 3910 volts and ≥ 58.8 Hz in ≤ 13 seconds, and subsequently achieves steady state voltage of ≥ 3910 and ≤ 4400 volts and frequency of ≥ 58.8 and ≤ 60.5 Hz.

5. Not Used.

6. Simulating a loss of offsite power in conjunction with an ESF actuation test signal, and:

   a) Verifying de-energization of the vital bus and load shedding from the vital bus.

   b) Verifying the diesel starts on the auto-start signal*, energizes the vital bus with permanently connected loads within 13 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 of these loads, the steady state voltage and frequency of the vital bus shall be maintained at ≥ 3910 and ≤ 4400 volts and ≥ 58.8 and ≤ 60.5 Hz during this test.

SURVEILLANCE REQUIREMENTS (Continued)

c) Verifying that all nonessential automatic diesel generator trips (i.e., other than engine overspeed, lube oil pressure low, 4 KV bus differential and generator differential), are automatically bypassed upon loss of voltage on the vital bus concurrent with a safety injection actuation signal.

7. Deleted

8. Verifying that the auto-connected loads to each diesel generator do not exceed the two hour rating of 2860 kw.

9. 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.

INSERT 1

e. ~~At least once per ten years~~ or after any modifications which could affect diesel generator interdependence by starting all diesel generators simultaneously*, during shutdown, and verifying that all diesel generators accelerate to at least 58.8 Hz in less than or equal to 13 seconds.

INSERT 1

f. ~~At least once per 18 months~~, the following test shall be performed within 5 minutes of diesel shutdown after the diesel has operated for at least two hours at 2340-2600 kw**:

Verifying the diesel generator starts and achieves ≥ 3910 volts and ≥ 58.8 Hz in ≤ 13 seconds, and subsequently achieves steady state voltage of ≥ 3910 and ≤ 4400 volts and frequency of 60 ± 1.2 Hz.

INSERT 1

g. ~~At least once per 18 months~~ verifying the diesel generator operates for at least 24 hours*. During the first 2 hours of this test, the diesel generators shall be loaded to 2760-2860 Kw**. During the remaining 22 hours of this test, the diesel generator shall be loaded to 2500-2600 Kw**. The steady state voltage and frequency shall be maintained at ≥ 3910 and ≤ 4580 volts and 60 ± 1.2 Hz during this test.

4.8.1.1.3 The diesel fuel oil storage ~~and transfer system shall be~~ demonstrated OPERABLE:

INSERT 1

a. ~~At least once per 31 days~~ by:

1. Verifying the level in each of the above required fuel storage tanks.

2. Verifying that both fuel transfer pumps can be started and transfer fuel from the fuel storage tanks to the day tanks.

INSERT 1

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that a sample of diesel fuel from each of the above required fuel storage tanks is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water and sediment.

4.8.1.1.4 Reports - NOT USED

* Surveillance testing may be conducted in accordance with the manufacturer's recommendations regarding engine prelube, warm-up and loading (unless loading times are specified in the individual Surveillance Requirements).

** This band is meant as guidance to preclude routine exceedances of the diesel generator manufacturer's design ratings. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A. C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generators:

4 kvolt	Vital Bus # 1A
4 kvolt	Vital Bus # 1B
4 kvolt	Vital Bus # 1C
460 volt	Vital Bus # 1A and associated control centers
460 volt	Vital Bus # 1B and associated control centers
460 volt	Vital Bus # 1C and associated control centers
230 volt	Vital Bus # 1A and associated control centers
230 volt	Vital Bus # 1B and associated control centers
230 volt	Vital Bus # 1C and associated control centers
115 volt	Vital Instrument Bus # 1A and Inverter *
115 volt	Vital Instrument Bus # 1B and Inverter *
115 volt	Vital Instrument Bus # 1C and Inverter *
115 volt	Vital Instrument Bus # 1D and inverter *

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With less than the above complement of A.C. busses OPERABLE or energized, restore the inoperable bus to OPERABLE and energized status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one inverter inoperable, energize the associated A. C. Vital Bus within 8 hours; restore the inoperable 1A, 1B, or 1C inverter to OPERABLE and energized 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; restore the inoperable 1D inverter to OPERABLE and energized 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.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators ~~at least once per 7 days~~ by verifying correct breaker alignment and indicated power availability.

(*) An inverter may be disconnected from its DC source for up to 24 hours for the purpose of performing an equalizing charge on its associated battery bank provided (1) its vital bus is OPERABLE and energized, and (2) the vital busses associated with the other battery banks are OPERABLE and energized.

ELECTRICAL POWER SYSTEMS

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two A.C. electrical bus trains shall be OPERABLE and energized from sources of power other than a diesel generator but aligned to an OPERABLE diesel generator with each train consisting of:

1 - 4 kvolt Vital Bus

1 - 460 volt Vital Bus and associated control centers

1 - 230 volt Vital Bus and associated control centers

1 - 115 volt Instrument Bus energized from its respective inverter connected to its respective D. C. bus train.

APPLICABILITY: MODES 5 and 6.
During movement of irradiated fuel assemblies.

ACTION:

With less than the above complement of A.C. busses and inverters OPERABLE and energized, immediately declare the affected required features inoperable, or suspend all operations involving CORE ALTERATIONS, positive reactivity changes, and movement of irradiated fuel assemblies until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses and inverters shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

INSERT 1

INSERT 1

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and power availability.

4.8.2.3.2 Each 125-volt battery and ~~above required~~ charger shall be demonstrated OPERABLE:

INSERT 1

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8.2.3-1 meet Category A limits.

2. The overall battery voltage is greater than or equal to 125 volts on float charge.

INSERT 1

b. At least once per 92 days and once within 24 hours after a battery discharge < 110 V and once within 24 hours after a battery overcharge > 150 V by verifying that the parameters in Table 4.8.2.3-1 meet the Category B limits.

INSERT 1c

c. At least once per 92 days by verifying that:

1. There is no visible corrosion at terminals or connectors or the connection resistance is:

≤150 micro ohms for inter-cell connections,
≤350 micro ohms for inter-rack connections,
≤350 micro ohms for inter-tier connections,
≤70 micro ohms for field cable terminal connections, and
≤2500 micro ohms for the total battery connection

resistance which includes all inter-cell connections (including bus bars), all inter-rack connections (including cable resistance) all inter-tier connections (including cable resistance) and all field terminal connections at the battery.

INSERT 1

2. The average electrolyte temperature of the representative cells is above 65°F.

d. At least once per 12 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

2. Remove visible terminal corrosion and verify cell-to-cell and terminal connections are coated with anti-corrosion material.

SURVEILLANCE REQUIREMENTS (Continued)

3. The connection resistance is:
   ≤ 150 micro ohms for inter-cell connections,
   ≤ 350 micro ohms for inter-rack connections,
   ≤ 350 micro ohms for inter-tier connections,
   ≤ 70 micro ohms for field cable terminal connections, and
   ≤ 2500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-rack connections (including cable resistance), all inter-tier connections (including cable resistance), and all field terminal connections at the battery.

INSERT 1

e. ~~At least once per 18 months~~ by verifying that the battery charger will supply at least 170 amperes at 125 volts for at least 4 hours.

INSERT 1

f.* ~~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.

INSERT 1

g. ~~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. Satisfactory completion of this performance discharge test shall also satisfy the requirements of Specification 4.8.2.3.2.f if the performance discharge test is conducted during a shutdown where that test and the battery service test would both be required.

h. At least once per 12 months, during shutdown, if the battery shows signs of degradation OR has reached 85% of the service life with a capacity less than 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its capacity on the previous performance test, or is below 90% of the manufacturer's rating.

i. At least once per 24 months, during shutdown, if the battery has reached 85% of the service life with capacity greater than or equal to 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test.

* A one-time extension to this surveillance requirement is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.

Table 4.8.2.3-1

Battery Cell Parameters Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE VALUE FOR EACH CONNECTED CELL
Electrolyte level	>Minimum level indication mark, and ≤1/4 above maximum level indication mark (a)	>Minimum level indication mark, and ≤1/4 above maximum level indication mark (a)	Above top of plates, and not overflowing
Float Voltage	≥2.13 volts	≥2.13 volts	≥2.07 volts
Specific Gravity (b) (c)	≥1.195	≥1.190 AND Average of all connected cells ≥1.200	Not more than 0.020 below average of all connected cells AND Average of all connected cells ≥1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charge provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 3 amps when on float charge.

(c) Or battery charging current is < 3 amps when on float charge. This is acceptable only during a maximum of 7 days following a battery recharge.

ELECTRICAL POWER SYSTEMS

125-VOLT D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

2 - 125-volt D.C. busses, and

2 - 125-volt batteries, each with at least one full capacity charger, associated with each of the above D.C. busses.

APPLICABILITY: MODES 5 and 6.
During movement of irradiated fuel assemblies.

ACTION:

With less than the above complement of D.C. equipment and busses OPERABLE, immediately declare the affected required features inoperable, or suspend all operations involving CORE ALTERATIONS, positive reactivity changes, and movement of irradiated fuel assemblies until the minimum required 125Volt D.C. electrical power sources are restored ~~to OPERABLE status.~~

INSERT 1

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. busses shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt batteries and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.8.2.5.1 Each D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and power availability.

4.8.2.5.2 Each 28-volt battery and above required charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

INSERT 1

1. The parameters in Table 4.8.2.5-1 meet Category A limits.

2. The overall battery voltage is greater than or equal to 27 volts on float charge.

INSERT 1

b. At least once per 92 days and once within 24 hours after a battery discharge < 25.7 V and once within 24 hours after a battery overcharge > 35 V by verifying that the parameters in Table 4.8.2.5-1 meet the Category B limits,

c. At least once per 92 days by verifying that:

INSERT 1

1. There is no visible corrosion at terminals or connectors or the connection resistance is:

≤ 50 micro ohms for inter-cell connections,
≤ 200 micro ohms for inter-tier connections,
≤ 70 micro ohms for field cable terminal connections, and
≤ 500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-tier connections (including cable resistance) and all field terminal connections at the battery.

INSERT 1

2. The average electrolyte temperature of the representative cells is ≥ 65°F.

d. At least once per 12 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

2. Remove visible terminal corrosion and verify cell-to-cell and terminal connections are coated with anti-corrosion material.

3. The connection resistance is:

≤ 50 micro ohms for inter-cell connections,
≤ 200 micro ohms for inter-tier connections,
≤ 70 micro ohms for field cable terminal connections, and
≤ 500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-tier connections (including cable resistance) and all field terminal connections at the battery.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by verifying that the battery charger will supply ≥ 150 amperes at ≥ 28 volts for ≥ 4 hours.

INSERT 1

f. 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.

g. 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. Satisfactory completion of this performance discharge test shall also satisfy the requirements of Specification 4.8.2.5.2.f if the performance discharge test is conducted during a shutdown where that test and the battery service test would both be required.

h. At least once per 12 months, during shutdown, if the battery shows signs of degradation OR has reached 85% of the service life with a capacity less than 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its capacity on the previous performance test, or is below 90% of the manufacturer's rating.

i. At least once per 24 months, during shutdown, if the battery has reached 85% of the service life with capacity greater than or equal to 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test.

[INFORMATION ONLY]

TABLE 4.8.2.5-1

BATTERY CELL PARAMETER REQUIREMENTS

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark and ≤ 1/4 inch above maximum level indication mark[a]	>Minimum level indication mark and ≤ 1/4 inch above maximum level indication mark[a]	Above top of plates and not overflowing
Float Voltage	≥2.13 V	≥2.13 V	≥2.07 V
Specific Gravity[b][c]	≥1.195	≥1.190 AND Average of all Connected cells ≥1.200	Not more than 0.020 below the average of all connected cells AND Average of all connected cells ≥1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charge provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) Or battery charging current is < 2 amps when on float charge. This is acceptable only during a maximum of 7 days following a battery recharge.

ELECTRICAL POWER SYSTEMS

28-VOLT D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.6 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

1 - 28-volt D.C. bus, and

1 - 28-volt battery and at least one full capacity charger associated with the above D.C. bus.

APPLICABILITY: MODES 5 and 6.
During movement of irradiated fuel assemblies.

ACTION:

With less than the above complement of D.C. equipment and busses OPERABLE, immediately declare the affected required features inoperable, or suspend all operations involving CORE ALTERATIONS, positive reactivity changes, and movement or irradiated fuel assemblies until the minimum required 28Volt D.C. electrical power sources are restored to OPERABLE status.

INSERT 7

SURVEILLANCE REQUIREMENTS

4.8.2.6.1 The above required 28-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the bus.

4.8.2.6.2 The above required 28-volt batteries and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.5.2.

3/4 8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.3.1 All containment penetration conductor overcurrent protective devices required to provide thermal protection of penetrations shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:
With one or more of the required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping either the primary or backup protective device, or racking out or removing the primary or backup device within 72 hours, declare the affected system or component inoperable, and verify the primary or backup protective device to be tripped, or the primary or backup device racked out or removed at least once per 7 days thereafter; or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 All required containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

INSERT 1

a. At least once per 18 months:

5

1.*,** For at least one 4.16 KV reactor coolant pump circuit, such that all reactor coolant pump circuits are demonstrated OPERABLE at least once per 72 months, by performance of:

(a) A CHANNEL CALIBRATION of the associated protective relays, and

(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.

* A one time extension to this surveillance requirement for inspection calibration and maggaring of the 1F 4KV Bus overload relays, which partially satisfies this surveillance requirement, is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.

** A one time extension to this surveillance requirement for inspection calibration and maggaring of the 1A, 1B, and 1C 480 transformer relays and CT's, which partially satisfy this surveillance requirement, is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.

SURVEILLANCE REQUIREMENTS (Continued)

2. By verifying the OPERABILITY of the required molded case and lower voltage circuit breakers, by selecting and functionally testing a representative sample of at least 10% of all the circuit breakers of that type. Circuit breakers selected for functional testing shall be selected on a rotating basis. The functional test shall consist of injecting a current input at the specified setpoint to each selected circuit breaker and verifying that each circuit breaker functions as designed. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during the 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.

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.

INSERT 1

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of the Reactor Coolant System, the fuel storage pool, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 6 (Only applicable to the refueling canal, the fuel storage pool and refueling cavity when connected to the Reactor Coolant System).

ACTION:
With the requirements of the above specification not satisfied, immediately

a. Suspend CORE ALTERATIONS and
b. Suspend positive reactivity additions and
c. Initiate action to restore boron concentration to within limit specified in the COLR.
d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1. Verify the boron concentration is within the limit of the COLR every 72 hours.

INSERT 1

REFUELING OPERATIONS

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

b. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

INSERT 1

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch inside door is capable of being closed and held in place by a minimum of four bolts, or an equivalent closure device installed and capable of being closed,

b. A minimum of one door in each airlock is capable of being closed

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
2. capable of being closed by the Containment Purge and Pressure-Vacuum Relief Isolation System.

Note: Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment building penetrations shall be determined to be either in its required condition or capable of being closed by a manual or automatic containment isolation valve at least once per 7 days. INSERT 1

4.9.4.2 Once per refueling prior to the start of movement of irradiated fuel assemblies within the containment building, verify the capability to close, within 1 hour, the equipment hatch inside door or an equivalent closure device. Applicable only when the equipment hatch is open during movement of irradiated fuel in the containment building.

4.9.4.3 Verify, once per 18 months, each required containment purge isolation valve actuates to the isolation position on a manual actuation signal.

INSERT 1

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION ALL WATER LEVELS

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop 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 pressure vessel hot legs.

c. The provisions of Specification 3.0.3 are not applicable.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.9.8.1 ~~At least once per 12 hours~~ one RHR loop shall be verified in operation and circulating coolant at a flow rate of:

a. greater than or equal to 1000 gpm, and
b. sufficient to maintain the RCS temperature at less than or equal to 140°F.

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movements of fuel assemblies or control rods.

INSERT 1

REFUELING OPERATIONS

STORAGE POOL WATER LEVEL

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:

With the requirements of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore water level to within its limit within 4 hours. The provisions of Specification 3.0.3 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.

INSERT 1

FUEL HANDLING AREA VENTILATION SYSTEM
LIMITING CONDITION FOR OPERATION

3.9.12 The Fuel Handling Area Ventilation System shall be OPERABLE with:

a. Two exhaust fans and one supply fan OPERABLE and operating, and
b. Capable of maintaining slightly negative pressure in the Fuel Handling Building.

APPLICABILITY: During movement of irradiated fuel within the Fuel Handling Building

ACTION:

a. With no Fuel Handling Area Ventilation System OPERABLE, suspend all operations involving movement of fuel within the storage pool until the Fuel Handling Area Ventilation System is restored to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required ventilation system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that, the Fuel Handling Building is maintained at a slightly negative pressure with respect to atmospheric pressure.

INSERT 1

b. At least once per 31 days by verifying both exhaust fans and one supply fan start and operate for at least 15 minutes, if not operating already.

c. At least once per 18 months by verifying a system flowrate of 19,490 cfm ± 10% during system operation.

3/4.10 SPECIAL TEST EXCEPTIONS

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 the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length rod either partially or FULLY WITHDRAWN shall be determined at least once per 2 hours. INSERT 1

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

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, 3.1.3.4, 3.1.3.5, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained ≤ 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 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

INSERT 1

4.10.2.1 The THERMAL POWER shall be determined to be ≤ 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The below listed surveillance requirements shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Surveillances 4.2.2.2 and 4.2.2.3.

b. Surveillances 4.2.3.1 and 4.2.3.2.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.4, and 3.1.3.5 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at ≤ 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be ≤ 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 a CHANNEL FUNCTIONAL TEST prior to initiating PHYSICS TESTS.

SPECIAL TEST EXCEPTION

NO FLOW TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set ≤ 25% of RATED THERMAL POWER

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint ~~at least once per hour~~ during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST prior to initiating startup or PHYSICS TESTS.

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outdoor 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 above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each outdoor temporary tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents ~~at least once per 7 days~~ when radioactive materials are being added to the tank.

INSERT 1

* Tanks included in this Specification are those outdoor temporary 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.

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume.

APPLICABILITY: At all times. *

ACTION:

a. With the concentration of oxygen in the waste gas holdup 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 waste gas holdup system greater than 4% by volume immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously** monitoring the waste gases in the waste gas holdup system with the oxygen monitor. If hydrogen is not measured, the concentration of hydrogen shall be assumed to exceed 4% by volume.

* Not applicable to portions of the Waste Gas System removed from service for maintenance provided that, the portions removed for maintenance are isolated, and purged of hydrogen to less than 4% by volume.

** If the oxygen monitoring instrumentation is inoperable, operation of the waste gas holdup system may continue provided grab samples are collected at least once per 24 hours and analyzed within the following 4 hours.

INSERT 1

In lieu of the above, the following inspection criteria are applicable only for Refueling Outage 18 and the subsequent operating cycle:
The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube beginning 17 inches below the top of the tubesheet on the tube hot leg side to 17 inches below the top of the tubesheet on the tube cold leg side.

The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 120, 90, and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary leakage.

INSERT 3

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

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, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1.3% delta k/k, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 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 increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s). INSERT 1

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control banks are within the limits in the COLR per Specification 3.1.3.5.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits in the COLR per Specification 3.1.3.5.

*See Special Test Exception 3.10.1

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit in the COLR per Specification 3.1.3.5.

INSERT 1

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

1. Reactor coolant system boron concentration,

2. Control rod position,

3. Reactor coolant system average temperature,

4. Fuel burnup based on gross thermal energy generation,

5. Xenon concentration, and

6. Samarium concentration.

INSERT 1

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

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% delta 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 SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

INSERT 1

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.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - 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 tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System if the boric acid storage system is OPERABLE, per Specification 3.1.2.6a while in MODE 4, or per Specification 3.1.2.5a while in MODE 5 or 6, 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 is OPERABLE per Specification 3.1.2.6b while in MODE 4, or per Specification 3.1.2.5b while in MODE 5 or 6.

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

a. When the boric acid tank is a required water source, by verifying at least once per 7 days that:

(1) The flow path from the boric acid tank to the boric acid transfer pump, the boric acid transfer pump, and the recirculation path from the boric acid transfer pump to the boric acid tank is ≥ 63°F, and

(2) The flow path between the boric acid transfer pump recirculation line to the charging pump suction line is ≥ 50°F,

INSERT 1

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 three boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two 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 Each of the above required flow paths shall be demonstrated OPERABLE:

a. By verifying at least once per 7 days that:

(1) The flow path from the boric acid tank to the boric acid transfer pump and from the recirculation line back to the boric acid tank is ≥ 63°F, and

INSERT 1

(2) The flow path between the boric acid tank recirculation line to the charging pump suction line is ≥ 50°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 during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 33 gpm to the Reactor Coolant System.

BORATED WATER SOURCES - 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 system with:

1. A minimum contained volume of 2,600 gallons,
2. Between 6,560 and 6,990 ppm of boron, and
3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 37,000 gallons,
2. A minimum boron concentration of 2,300 ppm, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

a. For the boric acid storage system, when it is the source of borated water ~~at least once per 7 days~~ by:

INSERT 1

1. Verifying the boron concentration of the water,
2. Verifying the water level of the tank, and
3. Verifying the boric acid storage tank solution temperature when it is the source of ~~borated water~~.

INSERT 1

b. For the refueling water storage tank by:

1. Verifying the boron concentration ~~at least once per 7 days~~,
2. Verifying the borated water volume ~~at least once per 7 days~~, and
3. Verifying the solution temperature ~~at least once per 24 hours~~ when it is the source of borated water and the outside air temperature is less than 35°F.

REACTIVITY CONTROL SYSTEMS

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 Specifications 3.1.2.1 and 3.1.2.2:

a. A boric acid storage system with:
   1. A contained volume of borated water in accordance with figure 3.1-2,
   2. A Boron concentration in accordance with Figure 3.1-2, and
   3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:
   1. A contained volume of between 364,500 and 400,000 gallons of water,
   2. A boron concentration of between 2,300 and 2,500 ppm, and
   3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. For the boric acid storage system, when it is the source of borated water at least once per 7 days by:
   1. Verifying the boron concentration in each water source.
   2. Verifying the water level of each water source, and
   3. Verifying the boric acid storage system solution temperature.

INSERT 1

b. For the refueling water storage tank by:
   1. Verifying the boron concentration at least once per 7 days,
   2. Verifying the borated water volume at least once per 7 days, and
   3. Verifying the solution temperature at least once per 24 hour when the outside air temperature is less than 35°F.

INSERT 1

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.

c) A core power distribution measurement is obtained and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.c.3.a and 3.1.3.1.c.3.c above are demonstrated.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the limits established in the limiting condition for operation at least once per 12 hours (allowing for one hour thermal soak after rod motion) except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

INSERT 1

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

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

1. Verify that all analog 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 18 steps when reactor power is ≤ 85% RATED THERMAL POWER or if reactor power is > 85% RATED THERMAL POWER, 12 steps of each other at least once per 8 hours, or

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

SURVEILLANCE REQUIREMENTS

INSERT 1

==========

4.1.3.2.1.1 Each analog rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 18 steps when reactor power is ≤ 85% RATED THERMAL POWER or if reactor power is > 85% RATED THERMAL POWER, 12 steps (allowing for one hour thermal soak after rod motion) ~~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 rod position indication system at least once per 4 hours.

4.1.3.2.1.2 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of a CHANNEL calibration ~~at least once per 18 months~~.

INSERT 1

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 230 steps withdrawn shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

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

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 & 2.

ACTION:

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

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 76% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

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

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

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. ~~At least once per 18 months.~~

INSERT 1

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be FULLY WITHDRAWN.

APPLICABILITY: MODES 1*, and 2*#@.

ACTION:

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

a. FULLY WITHDRAW the rod, or,

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

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be FULLY WITHDRAWN by use of the group demand counters, and verified by the analog rod position indicators**:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, and D during an approach to reactor critically, and

b. At least once per 12 hours thereafter.

INSERT 1

* See Special Test Exceptions 3.10.2 and 3.10.3.

** For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

@ Surveillance 4.1.3.4.a is applicable prior to withdrawing any control banks in preparation for startup (Mode 2).

\# With Keff greater than or equal to 1.0.

Note: This page effective prior to startup from fifth refueling outage scheduled to begin March 1990. Letter dated Jan. 11, 1990.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1*, and 2*#

ACTION:

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

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

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the CLOR, or

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

INSERT 1

SURVEILLANCE REQUIREMENTS

4.1.3.5 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours by use of the group demand counters and verified by ~~the analog rod~~ position indicators** 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 3.10.2 and 3.10.3

** For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

\# With Keff greater than or equal to 1.0

LIMITING CONDITION FOR OPERATION (Continued)

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

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

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE 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:

INSERT 1

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

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

INSERT 1

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

INSERT 1

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

INSERT 1

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.

INSERT 1

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 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.

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:

1. When THERMAL POWER is ≤ 25%, but > 5% of RATED THERMAL POWER, or

2. When the Power Distribution Monitoring System (PDMS) is inoperable;

and increasing the Measured $F_Q(Z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS or the moveable incore detectors when THERMAL POWER is > 25% of RATED THERMAL POWER, and increasing the measured $F_Q(Z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

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

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

2. The relationship:

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

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the power factor multiplier for F_{xy} in the COLR, 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 core power distribution measurements shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L:

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

b) At least once per 31 EFPD, whichever occurs first.

INSERT 1

SURVEILLANCE REQUIREMENTS (Continued)

2. When the F^C_{xy} is less than or equal to the F^{RTP}_{xy} limit for the appropriate measured core plane, additional core power distribution measurements shall be taken and F^C_{xy} compared to F^{RTP}_{xy} and F^L_{xy} ~~at least once per 31 EFPD~~. INSERT 1

e. The F_{xy} limit for Rated Thermal Power (F^{RTP}_{xy}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in the COLR per specification 6.9.1.9.

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

1. Lower core region from 0% to 15%, inclusive.
2. Upper core region from 85% to 100%, inclusive.
3. Grid plane regions at 17.8% ± 2%, 32.1% ± 2%, 46.4% ± 2%, 60.6% ± 2% and 74.9% ± 2%, inclusive.
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever F^C_{xy} exceeds F^L_{xy}.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a core power distribution measurement and increased by the applicable manufacturing and measurement uncertainties* as specified in the COLR.

* For Cycle 11, when the number of available movable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 5% measurement uncertainty shall be increased to [5% + (3-T/14.5)(1%)] where T is the number of available thimbles.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by obtaining a core power distribution measurement:

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.

INSERT 1

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by the applicable $F^N_{\Delta H}$ uncertainties* specified in the COLR.

* For Cycle 11, when the number of available movable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 4% measurement uncertainty shall be increased to [4% + (3-T/14.5)(1%)] where T is the number of available thimbles.

LIMITING CONDITION FOR OPERATION (Continued)

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

   (a) Either 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 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.

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.

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:

INSERT 1

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

INSERT 1

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 obtaining a core power distribution measurement* to confirm that the normalized symmetric power distribution is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

INSERT 1

* Using either the movable incore detectors in the four pairs of symmetric thimble locations or the power distribution monitoring system.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure.

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.2.5.2 The Reactor Coolant System Total Flow Rate shall be determined to be within the limits of Table 3.2-1 by performing a precision heat balance within 24 hours after achieving steady state conditions ≥90% RATED THERMAL POWER at ~~least once per 18 months~~. The provisions of Specification 4.0.4 are not applicable.

INSERT 1

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

Specified in the Surveillance Frequency Control Program unless otherwise noted

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

INSERT 1

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be verified to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. Each verification shall include at least one logic train such that both logic trains are verified at least once per 36 months and one channel per function such that all channels are verified 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 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (15)	CHANNEL CALIBRATION (15)	CHANNEL FUNCTIONAL TEST (15)	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip Switch	N.A.	N.A.	~~R~~(9)	1, 2, and *
2. Power Range, Neutron Flux	~~S~~	~~D~~(2), ~~M~~(3) ~~and Q~~(6)	~~Q~~	1, 2, and 3*
3. Power Range, Neutron Flux, High Positive Rate	N.A.	~~R~~(6)	~~Q~~	1, 2
4. Deleted				
5. Intermediate Range, Neutron Flux	S	~~R~~(6)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	~~S~~(7)	~~R~~(6)	~~Q~~ and S/U(1) (16)	2, 3, 4, 5 and *
7. Overtemperature ΔT	~~S~~	~~R~~	~~Q~~	1, 2
8. Overpower ΔT	~~S~~	~~R~~	~~Q~~	1, 2
9. Pressurizer Pressure--Low	~~S~~	~~R~~	~~Q~~	1, 2
10. Pressurizer Pressure--High	~~S~~	~~R~~	~~Q~~	1, 2
11. Pressurizer Water Level--High	~~S~~	~~R~~	~~Q~~	1, 2
12. Loss of Flow - Single Loop	~~S~~	~~R~~	~~Q~~	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (15)	CHANNEL CALIBRATION (15)	CHANNEL FUNCTIONAL TEST (15)	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow Two Loops	~~S~~	~~R~~	N.A.	1
14. Steam Generator Water Level--Low-Low	~~S~~	~~R~~	~~Q~~	1, 2
15. DELETED				
16. Undervoltage - Reactor Coolant Pumps	N.A.	~~R~~	~~Q~~	1
17. Underfrequency - Reactor Coolant Pumps	N.A.	~~R~~	~~Q~~	1
18. Turbine Trip				
a. Low Autostop Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
b. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	~~M~~(4) (5)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	~~R~~	1
21. Reactor Trip Breaker	N.A.	N.A.	~~M~~(5) (11) (13) and ~~R~~(14)	1, 2 and *
22. Automatic Trip Logic	N.A.	N.A.	~~M~~(5)	1, 2 and *

TABLE 4.3-1 (Continued)

NOTATION

* With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference ≥ 3 percent.

INSERT 1

(4) - Manual SSPS functional input check ~~every 18 months~~.

(5) - Each train or logic channel shall be tested ~~at least every 62 days on a STAGGERED TEST BASIS~~

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

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - Deleted

(9) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip mechanism for the Manual Reactor Trip Function.

The Test shall also verify OPERABILITY of the Bypass Breaker Trip circuits.

(10) - DELETED

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Reactor Trip Breaker Undervoltage and Shunt Trip mechanisms.

(12) - DELETED

TABLE 4.3-1 (Continued)

NOTATION

(13) - Verify operation of Bypass Breakers Shunt Trip function from local pushbutton while breaker is in the test position prior to placing breaker in service.

(14) - Perform a functional test of the Bypass Breakers U.V. Attachment via the SSPS.

(15) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

(16) - At the frequency specified in the Surveillance Frequency Control Program

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

specified in the Surveillance Frequency Control Program unless otherwise noted

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

INSERT 1

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit at least once per 18 months. Each verification shall include at least one logic train such that both logic trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3. The provisions of Specification 4.0.4 are not applicable to MSIV closure time testing. The provisions of Specification 4.0.4 are not applicable to the turbine driven auxiliary feedwater pump provided the surveillance is performed within 24 hours after the secondary steam generator pressure is greater than 680 psig.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (7)	CHANNEL CALIBRATION (7)	CHANNEL FUNCTIONAL TEST (7)	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION				
a. Manual Initiation	N.A.	N.A.	R	1,2,3,4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1,2,3,4
c. Containment Pressure--High	S	R	Q(3)	1,2,3
d. Pressurizer Pressure--Low	S	R	Q	1,2,3
e. Differential Pressure Between Steam Lines--High	S	R	Q	1,2,3
f. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low or Steam Line Pressure--Low	S	R	Q	1,2,3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	R	1,2,3,4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1,2,3,4
c. Containment Pressure--High-High	S	R	Q(3)	1,2,3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (7)	CHANNEL CALIBRATION (7)	CHANNEL FUNCTIONAL TEST (7)	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONTAINMENT ISOLATION				
a. Phase "A" Isolation				
1) Manual	N.A.	N.A.	~~R~~ Q	1,2,3,4
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	~~M~~ Q(2)	1,2,3,4
b. Phase "B" Isolation				
1) Manual	N.A.	N.A.	~~R~~ Q	1,2,3,4
2) Automatic Actuation Logic	N.A.	N.A.	~~M~~ Q(2)	1,2,3,4
3) Containment Pressure--High-High	~~S~~	~~R~~	~~Q~~ Q(3)	1,2,3
c. Containment Ventilation Isolation				
1) Manual	N.A.	N.A.	~~R~~ Q	1,2,3,4
2) Automatic Actuation Logic	N.A.	N.A.	~~M~~ Q(2)	1,2,3,4
3) Containment Atmosphere Gaseous Radioactivity-High		Per ~~table 4.3-3~~ Surveillance Requirement 4.3.3.1		

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (7)	CHANNEL CALIBRATION (7)	CHANNEL FUNCTIONAL TEST (7)	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	~~R~~	1,2,3**
b. Automatic Actuation Logic	N.A.	N.A.	~~M~~(2)	1,2,3
c. Containment Pressure--High-High	~~S~~	~~R~~	~~Q~~(3)	1,2,3
d. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low or Steam Line Pressure--Low	~~S~~	~~R~~	~~Q~~	1,2,3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	~~S~~	~~R~~	~~Q~~	1,2,3
6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC) LOGIC				
a. Inputs	N.A.	N.A.	~~M~~(6)	1,2,3,4
b. Logic, Timing and Outputs *	N.A.	N.A.	~~M~~(1)	1,2,3,4
7. UNDERVOLTAGE, VITAL BUS				
a. Loss of Voltage	~~S~~	~~R~~	~~M~~	1,2,3
b. Sustained Degraded Voltage	~~S~~	~~R~~	~~M~~	1,2,3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (7)	CHANNEL CALIBRATION (7)	CHANNEL FUNCTIONAL TEST (7)	MODES IN WHICH SURVEILLANCE REQUIRED
8. AUXILIARY FEEDWATER				
a. Automatic Actuation Logic	N.A.	N.A.	~~M~~ Q(2)	1,2,3
b. NOT USED				
c. Steam Generator Water Level--Low-Low	~~S~~ Q	~~R~~ Q	~~Q~~ Q	1,2,3
d. Undervoltage - RCP	~~S~~ Q	~~R~~ Q	~~Q~~ Q	1,2
e. S.I.	See 1 above (All S.I. surveillance requirements)			
f. Trip of Main Feedwater Pumps	N.A.	N.A.	S/U(4)	1,2
g. Station Blackout	See 6 and 7 above (SEC and U/V Vital Bus)			
9. SEMIAUTOMATIC TRANSFER TO RECIRCULATION				
a. RWST Low Level	~~S~~ Q	~~R~~ Q	~~Q~~ Q	1,2,3
b. Automatic Initiation Logic	N.A.	N.A.	~~M~~ Q(2)	1,2,3,4

TABLE 4.3-2 (Continued)

TABLE NOTATION

* Outputs are up to, but not including, the Output Relays. INSERT 1

** The provisions of Specification of 4.0.4 are not applicable.

(1) Each logic channel shall be tested at least once per 62 days on a STAGGERED TEST BASIS. The CHANNEL FUNCTION TEST of each logic channel shall verify that its associated diesel generator automatic load sequence timer is OPERABLE with the interval between each load block within 1 second of its design interval.

(2) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS. INSERT 1

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

(4) If not performed in the previous 92 days.

(5) NOT USED

(6) Inputs from undervoltage, Vital Bus, shall be tested monthly. Inputs from Solid State Protection System, shall be tested every 62 days on a STAGGERED TEST BASIS.

INSERT 1

INSERT 1

(7) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

INSERT 1

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNELS CHECKS	SOURCE CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS					
a. Fuel Storage Area	S	M	R	Q	*
2. PROCESS MONITORS					
a. Containment Monitors					
1) Gaseous Activity					
a) Purge & Pressure Vacuum Relief Isolation	S	M	R	Q	1, 2, 3, 4 & 5
b) RCS Leakage Detection	S	M	R	Q	1, 2, 3 & 4
2) Air Particulate Activity					
a) (NOT USED)					
b) RCS Leakage Detection	S	M	R	Q	1, 2, 3 & 4

*With fuel in the storage pool or building.

DELETED

TABLE 4.3-3 (Continued)
RADIATION MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNELS CHECKS	SOURCE CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	S	M	R	Q	1, 2, 3 & 4
2) High Range Auxiliary Building Exhaust System (Plant Vent)	S	M	R	Q	1, 2, 3 & 4
3) Main Steamline Discharge (Safety Valves and Atmospheric Dumps)	S	M	R	Q	1, 2, 3 & 4
4) Condenser Exh. Sys.	S	M	R	Q	1, 2, 3 & 4
3. CONTROL ROOM					
a. Air Intake - Radiation Level	S	M	R	Q	**

** ALL MODES and during movement of irradiated fuel assemblies and during CORE ALTERATIONS.

(delete this page)

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

INSERT 1

DELETED

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Pressurizer Pressure	M	R
2. Pressurizer Level	M	R
3. Steam Generator Pressure	M	R
4. Steam Generator Level	M	R

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be operable.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. As shown in Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-11.

specified in the Surveillance Frequency Control Program unless otherwise noted

TABLE 4.3-11
SURVEILLANCE REQUIREMENTS FOR ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECKS (1)	CHANNEL CALIBRATION (1)	CHANNEL FUNCTIONAL TEST (1)
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	~~M~~	~~R~~	N.A.
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	~~M~~	~~R~~	N.A.
3. Reactor Coolant Pressure (Wide Range)	~~M~~	~~R~~	N.A.
4. Pressurizer Water Level	~~M~~	~~R~~	N.A.
5. Steam Line Pressure	~~M~~	~~R~~	N.A.
6. Steam Generator Water Level (Narrow Range)	~~M~~	~~R~~	N.A.
7. Steam Generator Water Level (Wide Range)	~~M~~	~~R~~	N.A.
8. Refueling Water Storage Tank Water Level	~~M~~	~~R~~	N.A.
9. deleted			
10. Auxiliary Feedwater Flow Rate	S/U#	~~R~~	N.A.
11. Reactor Coolant System Subcooling Margin Monitor	~~M~~	N.A.*	N.A.

INSERT 1

#Auxiliary Feedwater System is used on each startup and flow rate indication is verified at that time.

*The instruments used to develop RCS subcooling margin are calibrated on an 18 month cycle; the monitor will be compared quarterly with calculated subcooling margin for known input values.

Insert 1

TABLE 4.3-11 (Continued)
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECKS (1)	CHANNEL CALIBRATION (1)	CHANNEL FUNCTIONAL TEST (1)
12. PORV Position Indicator	M	N.A.	R
13. PORV Block Valve Position Indicator	M	N.A.	Q*
14. Pressurizer Safety Valve Position Indicator	M	N.A.	R
15. Containment Pressure - Narrow Range	M	R	N.A.
16. Containment Pressure - Wide Range	M	R	N.A.
17. Containment Water Level - Wide Range	M	R	N.A.
18. Core Exit Thermocouples	M	R	N.A.
19. Reactor Vessel Level Instrumentation System (RVLIS)	M	R	N.A.
20. Containment High Range Accident Radiation monitor	S	R	Q
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor	S	R	Q

* Unless the block valve is closed in order to meet the requirements of Action b, or c in specification 3.4.5.

Table Notation

(1) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE to ensure that the limits of ODCM Control 3.11.1.1 are not exceeded.

APPLICABILITY: At all times.

ACTION:

a. Not Used

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Exert best efforts to return the instrument to OPERABLE status within 30 days and, if unsuccessful, explain in the next annual radioactive effluent release report why the inoperability was not corrected in a timely manner.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies ~~shown~~ in Table 4.3-12.

specified in the Surveillance Frequency Control Program unless otherwise noted

TABLE 4.3-12
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK (1)	SOURCE CHECK (1)	CHANNEL CALIBRATION (1)	CHANNEL FUNCTIONAL TEST (1)
1. Not Used				
2. Not Used				
3. Not Used				
4. TANK LEVEL INDICATING DEVICES**				
a. Temporary Outside Storage Tanks as Required	D*	N.A.	~~R~~	~~Q~~

TABLE NOTATION

* During liquid additions to the tank.

** If tank level indication is not provided, verification will be done by visual inspection.

(1) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILTY.- MODE 1, above 25% RATED THERMAL POWER (RTP)

ACTION:

With any of the operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, or 3.3.3.14.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system, satisfying the OPERABILITY requirements listed in Specification 3.3.3.2, to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.14.1 The operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, and 3.3.3.14.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

INSERT 1

4.3.3.14.2 Calibration of the PDMS is required:

a. At least once every 180 Effective Full Power Days when the minimum number and core coverage criteria as defined in 3.3.3.14.b.1 and 3.3.3.14.b.2 are satisfied, or

INSERT 1

b. At least once every 31 Effective Full Power Days when only the minimum number criterion as defined in 3.3.3.14.b.3 is satisfied.

3/4.4 REACTOR COOLANT SYSTEM

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 coolant loops in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENT

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

INSERT 1

*See Special Test Exception 3.10.4.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.4.1.2.1 At least the above required ~~reactor coolant~~ pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

INSERT 1

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 5% (narrow range) at least once per 12 hours.

INSERT 1

*All reactor coolant pumps may be de-energized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration (2) core outlet temperature is maintained at least 10°F below saturation temperature, and (3) the rod control system is de-energized**

**The rod control system shall be considered de-energized when one or more of the following conditions exist:

1) Both Rod Drive MG set motor breakers are open.

2) Both Rod Drive MG set generator breakers are open.

3) A combination of at least three of the Reactor Trip and/or Reactor Trip Bypass Breakers are open.

If none of the above conditions for de-energizing the rod control system are met, the system shall be considered energized.

SURVEILLANCE REQUIREMENTS

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability. INSERT 1

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 5% (narrow range) at least once per 12 hours. INSERT 1

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. INSERT 1

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.4 Two# residual heat removal loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5.##

ACTION:

a. With less than the above required loops operable, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no RHR 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 RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at ~~least once~~ per ~~12~~ hours.

INSERT 1

One RHR loop may be inoperable for up to ~~two hours for surveillance~~ testing, provided the other RHR loop is OPERABLE and in operation. Additionally, four filled reactor coolant loops, with at least two steam generators with their secondary side water levels greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop.

A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 312°F unless 1) the pressurizer water volume is less than 1650 cubic feet (equivalent to approximately 92% of level), or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

* Systems supporting RHR loop operability may be excepted as follows:

a. The normal or emergency power source may be inoperable.

** The residual heat removal pumps may be de-energized for up to 2 hours 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.

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1650 cubic feet (92% indicated level), and at least two groups of pressurizer heaters each having a capacity of ≥ 150 kw and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

INSERT 1

4.4.4.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring circuit current at least once each refueling outage.

4.4.4.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.

3/4.4.5 RELIEF VALVES

SURVEILLANCE REQUIREMENTS

INSERT 1

4.4.5.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

b. Operating solenoid valves, air control valves, and check valves on associated air accumulators in PORV control systems through one complete cycle of full travel, and

c. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

INSERT 1

4.4.5.2 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 b, or c in Specification 3.4.5.

3/4.4.7 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.7.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. The containment atmosphere particulate radioactivity monitoring system,

b. The containment pocket sump level monitoring system, and

c. Either the containment fan cooler condensate flow rate or the containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) monitoring systems-performance of CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

INSERT 1

b. Containment pocket sump level and containment fan cooler condensate flow rate (if being used) monitoring systems-performance of CHANNEL CALIBRATION at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. NOT USED

f. 1 GPM leakage at a Reactor Coolant System pressure of 2230 ±20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, or primary-to-secondary leakage not within limit, 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, and primary-to-secondary leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory at least once per 12 hours.

INSERT 1

c*. Verifying primary-to-secondary leakage is = 150 gallons per day through any one steam generator at least once per 72 hours during steady state operation.

INSERT 1

d*. Performance of a Reactor Coolant System water inventory balance** at least once per 72 hours. The water inventory balance shall be performed with the plant at steady state conditions. The provisions of specification 4.0.4 are not applicable for entry into Mode 4, and

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.7.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months.

INSERT 1

b. 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.

c. Prior to returning the valve to service following maintenance repair or replacement work on the valve.

b. For the Residual Heat Removal and Safety Injection Systems hot and cold leg injection valves and accumulator valves listed in Table 3.4-1 the testing will be done within 24 hours following valve actuation due to automatic or manual action or flow through the valve. For all other systems testing will be done once per refueling.

The provisions of specification 4.0.4 are not applicable for entry into MODE 3 or 4.

**Not required to be completed until 12 hours after establishment of steady state operation.*

***Not applicable to primary-to-secondary leakage.*

[INFORMATION ONLY]

REACTOR COOLANT SYSTEM

3/4.4.9 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.9 The specific activity of the primary coolant shall be limited to:

a. ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. ≤ 100/Ēμ Ci/gram.

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

ACTION:

MODES 1, 2 and 3*

a. With the specific activity of the primary coolant > 1.0 μCi/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} < 500°F within 6 hours.

b. With the specific activity of the primary coolant > 100/Ēμ Ci/gram, be in at least HOT STANDBY with T_{avg} < 500°F within 6 hours.

c. LCO 3.0.4.c is applicable.

MODES 1, 2, 3, 4 and 5

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > 100/Ēμ Ci/gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.9 The specific activity of the primary 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} ≥ 500°F.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

INSERT 1

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	~~At least once per 72 hours~~	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	~~1 per 14 days~~	1
3. Radiochemical for $\bar{E}$ 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 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

#Until the specific activity of the primary coolant system is restored within its limits.

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

REACTOR COOLANT SYSTEM

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.10.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

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

b. A maximum cooldown of 100°F in any one hour period, and

c. A maximum temperature change of less than or equal to 5°F in any one hour period, during hydrostatic testing operations above system design pressure.

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.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 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.

4.4.10.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.10.2 The pressurizer temperature shall be limited to:

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

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

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.10.2 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.

INSERT 1

[INFORMATION ONLY]

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.10.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two Pressurizer Overpressure Protection System relief valves (POPS) with a lift setting of less than or equal to 375 psig, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 3.14 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 312°F, except when the reactor vessel head is removed.

ACTION:

a. With one POPS inoperable in MODE 4 and the temperature of one or more of the RCS cold legs is less than or equal to 312°F, restore the inoperable POPS to OPERABLE status within 7 days or depressurize and vent the RCS through a 3.14 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

b. With one POPS inoperable in MODES 5 or 6 with the Reactor Vessel Head installed, restore the inoperable POPS to OPERABLE status within 24 hours, or complete depressurization and venting of the RCS through at least a 3.14 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

c. With both POPSs inoperable, depressurize and vent the RCS through a 3.14 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both POPSs have been restored to OPERABLE status.

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

e. LCO 3.0.4.b is not applicable when entering MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.10.3.1 Each POPS shall be demonstrated OPERABLE by:

SURVEILLANCE REQUIREMENTS (Continued)

a. Performance of a CHANNEL FUNCTIONAL TEST on the POPS actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the POPS is required OPERABLE and at least once per 31 days thereafter when the POPS is required OPERABLE.

INSERT 1

b. Performance of a CHANNEL CALIBRATION on the POPS actuation channel at least once per 18 months.

c. Verifying the POPS isolation valve is open at least once per 72 hours when the POPS is being used for overpressure protection.

d. Testing pursuant to Specification 4.0.5.

INSERT 1

4.4.10.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

INSERT 1

3/4.4.12 HEAD VENTS

LIMITING CONDITION FOR OPERATION

3.4.12 Four reactor vessel head vent paths shall be operable with the vent paths closed. A vent path consists of at least two head vent valves in series, powered from vital sources, and associated flowpath.

APPLICABILITY: MODES 1, 2, 3 AND 4.

ACTION:

a. With one, two or three reactor vessel head vent path(s) inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path(s) is maintained closed with the valve actuators key locked in the closed position; restore the inoperable vent path(s) 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 four reactor vessel head vent paths inoperable; maintain the inoperable vent paths 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

4.4.12 Reactor vessel head vent system vent paths shall be demonstrated OPERABLE at least once per 18 months by:

INSERT 1

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each valve in the vent paths through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

3. Verifying flow through the reactor vessel head vent system vent path during venting.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained volume of between 6223 and 6500 gallons of borated water

c. A boron concentration of between 2200 and 2500 ppm, and

d. A nitrogen cover-pressure of between 595.5 and 647.5 psig.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve or boron concentration outside the required limits, restore the inoperable accumulator to OPERABLE status within 24 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within 24 hours and be in HOT SHUTDOWN within the next 12 hours.

c. With the boron concentration of one accumulator outside the required limits, restore the boron concentration to within the required limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than or equal to 1000 psig within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. ~~At least once per 12 hours~~ by: INSERT 1

1. Verifying the water level and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

* Pressurizer Pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

INSERT 1

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution.

c. At least once per 31 days when the RCS pressure is greater than 1000 psig by verifying that the power lockout switch is in lockout.

d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically upon receipt of a safety injection test signal.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. ~~At least once per 12 hours~~ by:

1. Verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. 2 SJ 69	a. RHR pump suction	a. open
b. 2 SJ 30	b. SI pump suction	b. open
c. 21 SJ 40	c. SI discharge to hot legs	c. closed
d. 22 SJ 40	d. SI discharge to hot legs	d. closed
e. 2 RH 26	e. RHR discharge to hot legs	e. closed
f. 21 SJ 49	f. RHR discharge to cold legs	f. open
g. 22 SJ 49	g. RHR discharge to cold legs	g. open
h. 2 CS 14#	h. Spray additive tank discharge	h. open
i. 2 SJ 135	i. SI discharge to cold legs	i. open
j. 2 SJ 67	j. SI recirc. line isolation	j. open
k. 2 SJ 68	k. SI recirc. line isolation	k. open

2. Verifying that the following valves are in the indicated positions:

Valve Number	Valve Function	Valve Position
a. 21 RH 18	a. RHR crosstie valve	a. Open
b. 22 RH 19	b. RHR crosstie valve	b. Open

INSERT 1

b. ~~At least once per 31 days~~ by:

1. 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.

2. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points.

#If inoperable, the applicable Technical Specification is 3.6.2.2.

SURVEILLANCE REQUIREMENTS (Continued)

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

INSERT 1

2. ~~At least once daily (24 hour consecutive period)~~ the areas affected within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

d. ~~At least once per 18 months~~ by:

1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. ~~At least once per 18 months~~, 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 each of the following pumps start automatically upon receipt of a safety injection test signal:

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

SURVEILLANCE REQUIREMENTS (Continued)

f. By verifying that each of the following pumps develops the indicated Total Dynamic Head (TDH) when tested at the test flow point pursuant to Specification 4.0.5:

1.	Centrifugal Charging pump	≥ 2338 psi TDH
2.	Safety Injection pump	≥ 1369 psi TDH
3.	Residual Heat Removal pump	≥ 165 psi TDH

g. By verifying the correct position of each of 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.

2. At least once per 18 months.

INSERT 1

HPSI System Valve Number	LPSI System Valve Number
21 SJ 16	21 SJ 138
22 SJ 16	22 SJ 138
23 SJ 16	23 SJ 138
24 SJ 16	24 SJ 138
	21 SJ 143
	22 SJ 143
	23 SJ 143
	24 SJ 143

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 Safety Injection pumps, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is ≥ 453 gpm, and

b) The total flow rate through all four injection lines is ≤ 647 gpm, and

c) The difference between any pair of injection line flow rates is ≤ 12.0 gpm, and

d) The total pump flow rate is ≤ 664 gpm in the cold leg alignment, and

e) The total pump flow rate is ≤ 654 gpm in the hot leg alignment.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All safety injection pumps and centrifugal charging pumps, except the above required OPERABLE pump, shall be demonstrated to be inoperable at least once per 12 hours while in MODE 4 and the temperature of one or more of the RCS cold legs is less than or equal to 312°F, MODE 5, or MODE 6 when the head is on the reactor vessel by either of the following methods:

a. By verifying that the motor circuit breakers have been removed from their electrical power supply circuits or,

b. By verifying that the pump is in a recirculation flow path and that two independent means of preventing RCS injection are utilized.

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SEAL INJECTION FLOW

LIMITING CONDITION FOR OPERATION

3.5.4 Reactor coolant pump seal injection flow shall be ≤40 gpm with centrifugal charging pump discharge header pressure ≥2430 psig and the charging flow control valve full open.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With seal injection flow not within the limit, adjust manual seal injection throttle valves to give a flow within the limit with the charging pump discharge pressure ≥2430 psig and the charging flow control valve full open within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.5.4 ~~At least once per 31 days~~, verify manual seal injection throttle valves are adjusted to give a flow within the limit with centrifugal charging pump discharge header pressure ≥2430 psig, and the charging flow control valve full open.

The provisions of Specification 4.0.4 are not applicable for entry into Mode 3. This exemption is allowed for up to 4 hours after the Reactor Coolant System pressure stabilizes at 2235 ± 20 psig.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A contained volume of between 364,500 and 400,000 gallons of borated water.

b. A boron concentration of between 2,300 and 2,500 ppm, and

c. A minimum water temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, 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.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. ~~At least once per 7 days~~ by: INSERT 1

1. Verifying the water level 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 < 35°F.

INSERT 1

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 one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

INSERT 1

a1. At least once per 31 days by verifying that each containment manual valve or blind flange that is located outside containment and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls. Valves and blind flanges in high radiation areas may be verified by use of administrative controls.

a2. Prior to entering Mode 4 from Mode 5 if not performed within the last 92 days by verifying that each containment manual valve or blind flange that is located inside containment and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls. Valves and blind flanges in high radiation areas may be verified by use of administrative controls.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. After each closing of a penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing in accordance with the Containment Leakage Rate Testing Program.

INSERT 1

d. At least once per 12 hours by verifying that the surveillance requirements of 4.6.2.3.a are met for penetrations associated with the containment fan coil units.

e. At least once per 18 months by verifying that the surveillance requirements of 4.6.2.3.d are met for penetrations associated with the containment fan coil units.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATIONS (Continued)

c. One or more containment air locks inoperable for reasons other than condition a. or b.

   1. Immediately initiate action to evaluate overall containment leakage per LCO 3.6.1, and:

   2. Verify that at least one door is closed in the affected air lock within 1 hour, and:

   3. Restore the air lock to OPERABLE status within 24 hours.

d. If the ACTIONS and associated completion times of a., b., or c. cannot be met, be in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying seal leakage rate in accordance with the Containment Leakage Rate Testing program.

b. By conducting an overall air lock leakage test in accordance with the Containment Leakage Rate Testing Program.

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

INSERT 1

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -1.5 and +0.3 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 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 shall be determined to be within the limits ~~at least once per 12 hours~~.

INSERT 1

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 Verify containment average air temperature is within limit at least once per twenty four hours.

INSERT 1

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 RHR pump discharge.

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.

INSERT 1

b. By verifying, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 204 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 High-High pressure test signal.

2. Verifying each spray pump starts automatically on a Containment High-High pressure test signal.

d. Following activities that could result in nozzle blockage, either evaluate the work performed to determine the impact to the containment spray system, or perform 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 2568 and 4000 gallons of between 30 and 32 percent by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the spray additive system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

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

b. At least once per 6 months by:

INSERT 1

1. Verifying the solution level 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 High-High pressure test signal.

d. At least once per 5 years by:

1. Verifying a NaOH solution flow rate of 12.0 ± 3.0 gpm from the spray additive tank through sample valve 2CS61 with the spray additive tank at 2.5 ± 0.5 psig and

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the spray additive tank eductor flow will be 35 ± 3.5 gpm to each containment spray system. Testing may be performed by measuring the flow of borated water from the RWST through the installed 2" test line and Valve CS31; using this test line up with the spray pump operating in the recirculation mode and the RWST level at 41 feet ± 0.5 feet, the measured flow shall be 57 gpm ± 5.7 gpm.

INFORMATION ONLY

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Five containment cooling fans shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or two of the above required containment cooling fans inoperable, restore the inoperable cooling fan(s) to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With three or more of the above required containment cooling fans inoperable, restore at least three cooling fans 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. Restore the remaining inoperable cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment cooling fan shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 12 hours by:

1. Verifying the water level in each service water accumulator vessel is greater than or equal to 226 inches and less than or equal to 252 inches.

2. Verifying the temperature in each service water accumulator vessel is greater than or equal to 55°F and less than or equal to 95°F.

3. Verifying the nitrogen cover pressure in each service water accumulator vessel is greater than or equal to 135 psig and less than or equal to 160 psig.

b. At least once per 31 days by:

INSERT 1

1. Starting (unless already operating) each fan from the control room in low speed.

2. Verifying that each fan operates for at least 15 minutes in low speed.

3. Verifying a cooling water flow rate of greater than or equal to 1300 gpm to each cooler.

c. At least once per 18 months by verifying that on a safety injection test signal:

1. Each fan starts automatically in low speed.

2. The automatic valves and dampers actuate to their correct positions and that the cooling water flow rate to each cooler is greater than or equal to 1300 gpm.

d. At least once per 18 months by verifying that on a loss of offsite power test signal, each service water accumulator vessel discharge valve response time is within limits.

SURVEILLANCE REQUIREMENTS (Continued)

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

INSERT 1

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

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. NOT USED

d. Verifying that on a Containment Purge and Pressure-Vacuum Relief isolation test signal, each required Purge and each Pressure-Vacuum Relief valve actuates to its isolation position.

INSERT 1

e. Verifying that the Containment Pressure-Vacuum Relief Isolation valves are limited to ≤ 60° opening angle.

4.6.3.3 At least once per 18 months, verify that on a main steam isolation test signal, each main steam isolation valve actuates to its isolation position.

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

INSERT 1

4.6.3.5 Each required containment purge isolation valve shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cyclings, then at least once per 72 hours, by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Specification 4.6.1.2.b for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60La.

4.6.3.6 A pressure drop test to identify excessive degradation of resilient valve seals shall be conducted on the:

a. Required Containment Purge Supply and Exhaust Isolation Valves at least once per 6 months.

b. Deleted.

INSERT 1

4.6.3.7 The required containment purge supply and exhaust isolation valves shall be determined closed at least once per 31 days.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated manual activation switches in the control room and flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate vital busses, and

b. One 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.

d. LCO 3.0.4.b is not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. ~~At least once per 31 days by:~~

INSERT 1

1. 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.

2. Verify the manual maintenance valves in the flow path to each steam generator are locked open.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days on a STAGGERED TEST BASIS by:

INSERT 1

1. Verify that the developed head of each motor driven pump at the flow test point is greater than or equal to the required developed head.

2. Verify that the developed head of the steam driven pump at the flow test point is greater than or equal to the required developed head when the steam generator pressure is >680 psig. The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 24 hours after secondary side pressure is greater than 680 psig.

c. At least once per 18 months by:

INSERT 1

1. Verifying that each auxiliary feedwater automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

2. Verifying that each auxiliary feedwater pump starts automatically on an actual or simulated actuation signal.

The provisions of Specification 4.0.4 are not applicable to the turbine driven auxiliary feedwater pump, provided the surveillance is performed within 24 hours after the secondary steam generator pressure is greater than 680 psig.

AUXILIARY FEED STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The auxiliary feed storage tank (AFST) shall be OPERABLE with a contained volume of at least 200,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the auxiliary feed storage tank inoperable, within 4 hours either:

a. Restore the AFST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of a demineralized water or a fire protection/domestic water storage tank as a backup supply to the auxiliary feedwater pumps and restore the auxiliary feed storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.7.1.3.1 The auxiliary feed storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the water level is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 A demineralized water storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the tank contains greater than or equal to 200,000 gallons of water and by verifying proper alignment of valves for taking suction from this tank when it is the supply source for the auxiliary feedwater pumps.

INSERT 1

4.7.1.3.3 A fire protection/domestic water storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the tank contains greater than or equal to 200,000 gallons of water and by verifying proper alignment of valves for taking suction from this tank when it is the supply source for the auxiliary feedwater pumps.

4.7.1.3.4 The Service Water System shall be demonstrated capable of providing a water supply to the Auxiliary Feedwater System at least once per 12 months by verifying that the required spool-piece is on-site.

INSERT 1

[INFORMATION ONLY]

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 μCi/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the secondary coolant system greater than 0.10 μCi/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 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a. 1 per 31 days whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.
	b. 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

INSERT 1

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the primary and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig ~~at least once per hour~~ when the temperature of either the primary or secondary coolant is less than 70°F.

INSERT 1

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.*

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component ~~cooling water loops shall~~ be demonstrated OPERABLE:

INSERT 1

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.

*The time for completion of repairs to number 22 component cooling water heat exchanger shall be extended from 0700 hours on November 23, 1982 to 1900 hours on November 23, 1982. If repairs are not completed by that time, the unit shall be placed in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two service water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

INSERT 1

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on Safeguards Initiation signal.

3/4.7.5 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.5 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Delaware River exceeds 10.5' Mean Sea Level USGS datum, at the service water intake structure.

APPLICABILITY: At all times.

ACTION:

a. With the water level at the service water intake structure above elevation 10.5' Mean Sea Level USGS datum, close all watertight doors within 2 hours.

b. With the water level at the service water intake structure above elevation 11.5' Mean Sea Level USGS datum, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The water level at the service water intake structure shall be determined to be within the limits by:

INSERT 1

a. Measurement at least once per 24 hours when the water level is below elevation 10.5' Mean Sea Level USGS datum, and

INSERT 1

b. Measurement at least once per 2 hours when the water level is equal to or above elevation 10.5' Mean Sea Level USGS datum.

SURVEILLANCE REQUIREMENTS

4.7.6.1 The control room emergency air conditioning system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train(s) and verifying that the train(s) operates with each fan operating for at least 15 minutes.

INSERT 1

b. At least once per 18 months or prior to return to service (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 charcoal adsorbers remove ≥99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the ventilation system at a flow rate of 8000 cfm ±10%.

2. Verifying that the HEPA filter banks remove ≥99% of the DOP when they are tested in-place while operating the ventilation system at a flow rate of 8000 cfm ±10%.

3. Verifying within 31 days after removal from the CREACS unit, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the CREACS unit, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and a relative humidity of 95%.

d. At least once per 18 months by: INSERT 1

1. Verifying that the pressure drop across the combined HEPA filter and charcoal adsorber bank is ≤3.5 inches water gauge while operating the ventilation system at a flow rate of 8000 cfm ± 10%.

2. Verifying that on a safety injection test signal or control room intake high radiation test signal, the system automatically actuates in the pressurization mode by opening the outside air supply and diverting air flow through the HEPA filter and charcoal adsorber bank.

3. Deleted.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying that on a manual actuation signal, the system will actuate to the required pressurization or recirculation operating mode.

5. Verify each CREACS train has the capability to remove the assumed heat load.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place while operating the filter system at a flow rate of 8000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal absorber bank by verifying that the charcoal absorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the filter system at a flow rate of 8000 cfm ± 10%.

4.7.6.2 Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program (Refer to T.S. 6.17).

[INFORMATION ONLY]

4.7.7 The above required Auxiliary Building Ventilation System shall be demonstrated OPERABLE:

a) At least once per 12 hours by verifying negative pressure in the Auxiliary Building.

INSERT 1

b) At least once per 31 days by starting each fan, from the control room, each fan operates for at least 15 minutes.

c) At least once per 18 months by verifying that the system starts following a Safety Injection Test Signal.

3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.8 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 microcuries 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:

   1. Either decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.1 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 microcuries per test sample.

4.7.8.2 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 six months for all sealed sources containing radioactive materials:

INSERT 1

[INFORMATION ONLY]

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

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 six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

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 and following repair or maintenance to the source or detector.

4.7.8.3 Reports - 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.

LIMITING CONDITION FOR OPERATION

ACTION: MODES 5 and 6 or during movement of irradiated fuel assemblies.*

a. With one chiller inoperable:

1. Remove the appropriate non-essential heat loads from the Chilled Water System within 4 hours and;

2. Restore the chiller to OPERABLE status within 14 days or;

3. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

b. With two chillers inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Align the control room emergency air conditioning system (CREACs) for single filtration operation using the Salem Unit 1 train within 4 hours and;

3. Restore at least one chiller to OPERABLE status within 72 hours or;

4. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

c. With one chilled water pump inoperable, restore the chilled water pump to OPERABLE status within 7 days or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.10 The chilled water loop which services the safety-related loads in the Auxiliary Building shall be demonstrated OPERABLE:

INSERT 1

a. At least once per 31 days by verifying that each manual valve in the chilled water system flow path servicing safety related components 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 actuates to its correct position on a Safeguards Initiation signal.

c. At least once per 92 days by verifying that each chillers starts and runs.

* During Modes 5 and 6 and during movement of irradiated fuel assemblies, chilled water components are not considered to be inoperable solely on the basis that the backup emergency power source, diesel generator, is inoperable.

3/4.7.11 FUEL STORAGE POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.11 The fuel storage pool boron concentration shall be ≥ 800 ppm

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool.

ACTION:

With fuel storage pool boron concentration not within limit:

a. Immediately suspend movement of fuel assemblies in the fuel storage pool and

b. Initiate action to:

1. immediately restore fuel storage pool boron concentration to within limit or

2. immediately perform a fuel storage pool verification.

c. The provisions of Specification 3.0.3 is not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11 Verify the fuel storage pool boron concentration is within limit every 7 days

INSERT 1

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Two physically independent A.C. circuits between the offsite transmission network and the onsite Class 1E distribution system (vital bus system) shall be:

INSERT 1

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, power availability, and

INSERT 1

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) vital bus supply from one 13/4 kv transformer to the other 13/4 kv transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

INSERT 1

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in its day tank.

2. Verifying the diesel generator starts from standby conditions* and achieves ≥ 3910 volts and ≥ 58.8 Hz in ≤ 13 seconds, and subsequently achieves steady state voltage of ≥ 3910 and ≤ 4400 volts and frequency of 60 ± 1.2 Hz.

   Subsequently, verifying the generator is synchronized with voltage maintained ≥ 3910 and ≤ 4580 volts, gradually loaded to 2340-2600 kw**, and operates at a load of 2340-2600 kw for greater than or equal to 60 minutes.

3. Verifying the diesel generator is aligned to provide standby power to the associated vital bus.

INSERT 1

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks.

INSERT 1

c. At least once per 6 months by verifying the diesel generator starts from standby conditions* and achieves ≥ 3910 volts and ≥ 58.8 Hz in ≤ 13 seconds, and subsequently achieves steady state voltage of ≥ 3910 and ≤ 4400 volts and frequency of 60 ± 1.2 Hz.

   The generator shall be synchronized to its emergency bus with voltage maintained ≥ 3910 and ≤ 4580 volts, loaded to 2340-2600** kw in less than or equal to 60 seconds, and operate at a load of 2340-2600 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.2, may also serve to concurrently meet those requirements.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months during shutdown by: INSERT 1

1. DELETED

2. Verifying that, on rejection of a load greater than or equal to 820 kw, the voltage and frequency are restored to ≥ 3910 and ≤ 4400 volts and 60 ± 1.2 Hz within 4 seconds, and subsequently achieves a steady state frequency of ≥ 58.8 and ≤ 60.5 Hz.

3. Simulating a loss of offsite power by itself, and:

   a) Verifying de-energization of the vital bus and load shedding from the vital bus.

   b) Verifying the diesel starts on the auto-start signal*, energizes the vital bus with permanently connected loads within 13 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 of these loads, the steady state voltage and frequency of the vital bus shall be maintained at ≥ 3910 and ≤ 4400 volts and ≥ 58.8 and ≤ 60.5 Hz during this test.

4. 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 diesel generator shall achieve ≥ 3910 volts and ≥ 58.8 Hz in ≤ 13 seconds, and subsequently achieves steady state voltage of ≥ 3910 and ≤ 4400 volts and frequency of ≥ 58.8 and ≤ 60.5 Hz.

5. Deleted

6. Simulating a loss of offsite power in conjunction with an ESF actuation test signal, and

   a) Verifying de-energization of the vital bus and load shedding from the vital bus.

   b) Verifying the diesel starts on the auto-start signal*, energizes the vital bus with permanently connected loads within 13 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 of these loads, the steady state voltage and frequency of the vital bus shall be maintained at ≥ 3910 and ≤ 4400 volts and ≥ 58.8 and ≤ 60.5 Hz during this test.

SURVEILLANCE REQUIREMENTS (Continued)

c) Verifying that all nonessential automatic diesel generator trips (i.e., other than engine overspeed, lube oil pressure low, 4 KV Bus differential and generator differential) are automatically bypassed upon loss of voltage on the vital bus concurrent with a safety injection actuation signal.

7. Deleted

8. Verifying that the auto-connected loads to each diesel generator do not exceed the two hour rating of 2860 kw.

9. 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.

INSERT 1

e. ~~At least once per 10 years~~ or after any modifications which could affect diesel generator interdependence by starting all diesel generators simultaneously*, during shutdown, and verifying that all diesel generators accelerate ~~to~~ at ~~least 58.8~~ Hz in less than or equal to 13 seconds.

INSERT 1

f. ~~At least once per 18 months~~ the following test shall be performed within 5 minutes of diesel shutdown after the diesel has operated for at least two hours at 2340-2600 kw**:

Verifying the diesel generator starts and achieves ≥ 3910 volts and ≥ 58.8 Hz in ≤ 13 seconds, and subsequently achieves steady state voltage of ≥ 3910 and ≤ 4400 volts and frequency of 60 ± 1.2 Hz.

INSERT 1

g. ~~At least once per 18 months~~ verifying ~~the diesel~~ generator operates for at least 24 hours*. During the first 2 hours of this test, the diesel generators shall be loaded to 2760-2860 Kw**. During the remaining 22 hours of this test, the diesel generator shall be loaded to 2500-2600 Kw**. The steady state voltage and frequency shall be maintained at ≥ 3910 and ≤ 4580 volts and 60 ± 1.2 Hz during this test.

4.8.1.1.3 The diesel fuel oil storage and transfer ~~system~~ shall be demonstrated OPERABLE:

INSERT 1

a. ~~At least once per 31 days~~ by:

1. Verifying the level in each of the above required fuel storage tanks.

2. Verifying that both fuel transfer pumps can be started and transfer fuel from the fuel storage tanks to the day tanks.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that a sample of diesel fuel from each of the above required fuel storage tanks is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water and sediment. INSERT 1

4.8.1.1.4 Reports - NOT USED

* Surveillance testing may be conducted in accordance with the manufacturer's recommendations regarding engine prelube, warm-up and loading (unless loading times are specified in the individual Surveillance Requirements).

** This band is meant as guidance to preclude routine exceedances of the diesel generator manufacturer's design ratings. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A. C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generators:

4 kvolt	Vital Bus # 2A
4 kvolt	Vital Bus # 2B
4 kvolt	Vital Bus # 2C
460 volt	Vital Bus # 2A and associated control centers
460 volt	Vital Bus # 2B and associated control centers
460 volt	Vital Bus # 2C and associated control centers
230 volt	Vital Bus # 2A and associated control centers
230 volt	Vital Bus # 2B and associated control centers
230 volt	Vital Bus # 2C and associated control centers
115 volt	Vital Instrument Bus # 2A and Inverter *
115 volt	Vital Instrument Bus # 2B and Inverter *
115 volt	Vital Instrument Bus # 2C and Inverter *
115 volt	Vital Instrument Bus # 2D and Inverter *

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With less than the above complement of A.C. busses OPERABLE or energized, restore the inoperable busses to OPERABLE and energized status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one inverter inoperable, energize the associated A.C. Vital Bus within 8 hours; restore the inoperable 2A, 2B, or 2C inverter to OPERABLE and energized 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; restore the inoperable 2D inverter to OPERABLE and energized 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

INSERT

4.8.2.1 The specified A.C. busses and inverters shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

---

*An inverter may be disconnected from its D.C. source for up to 24 hours for the purpose of performing an equalizing charge on its associated battery bank provided (1) its vital bus is OPERABLE and energized, and (2) the vital busses associated with the other battery banks are OPERABLE and energized.

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two A.C. electrical bus trains shall be OPERABLE and energized from sources of power other than a diesel generator but aligned to an OPERABLE diesel generator with each train consisting of:

1 - 4 kvolt Vital Bus

1 - 460 volt Vital Bus and associated control centers

1 - 230 volt Vital Bus and associated control centers

1 - 115 volt Instrument Bus energized from its respective inverter connected to its respective D. C. bus train.

APPLICABILITY: MODES 5 and 6.
During movement of irradiated fuel assemblies.

ACTION:

With less than the above complement of A.C. busses and inverters OPERABLE and energized, immediately declare the affected required features inoperable, or suspend all operations involving CORE ALTERATIONS, positive reactivity changes, and movement of irradiated fuel assemblies until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses and inverters shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

INSERT 1

INSERT 1

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the bus.

4.8.2.3.2 Each required 125-volt battery and charger shall be demonstrated OPERABLE:

INSERT 1

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8.2.3-1 meet Category A limits.

2. The overall battery voltage is greater than or equal to 125 volts on float charge.

INSERT 1

b. At least once per 92 days and once within 24 hours after a battery discharge < 110 V and once within 24 hours after a battery overcharge > 150 V by verifying that the parameters in Table 4.8.2.3-1 meet the Category B limits.

INSERT 1

c. At least once per 92 days by verifying that:

1. There is no visible corrosion at terminals or connectors or the connection resistance is:

≤150 micro ohms for inter-cell connections,
≤350 micro ohms for inter-rack connections,
≤350 micro ohms for inter-tier connections,
≤70 micro ohms for field cable terminal connections, and
≤2500 micro ohms for the total battery connection

resistance which includes all inter-cell connections (including bus bars), all inter-rack connections (including cable resistance) all inter-tier connections (including cable resistance) and all field terminal connections at the battery.

2. The average electrolyte temperature of the representative cells is above 65°F.

INSERT 1

d. At least once per 12 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

2. Remove visible terminal corrosion and verify cell-to-cell and terminal connections are coated with anti-corrosion material.

SURVEILLANCE REQUIREMENTS (Continued)

3. The connection resistance is:
≤150 micro ohms for inter-cell connections,
≤350 micro ohms for inter-rack connections,
≤350 micro ohms for inter-tier connections,
≤70 micro ohms for field cable terminal connections, and
≤2500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-rack connections (including cable resistance) all inter-tier connections (including cable resistance) and all field terminal connections at the battery.

INSERT 1

e. ~~At least once per 18 months~~ by verifying that the battery charger will supply at least 170 amperes at 125 volts for at least 4 hours.

INSERT 1

f. ~~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.

INSERT 1

g. ~~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. Satisfactory completion of this performance discharge test shall also satisfy the requirements of Specification 4.8.2.3.2.f if the performance discharge test is conducted during a shutdown where that test and the battery service test would both be required.

h. At least once per 12 months, during shutdown, if the battery shows signs of degradation <u>OR</u> has reached 85% of the service life with a capacity less than 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its capacity on the previous performance test, or is below 90% of the manufacturer's rating.

i. At least once per 24 months, during shutdown, if the battery has reached 85% of the service life with capacity greater than or equal to 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test.

[INFORMATION ONLY]

SURVEILLANCE REQUIREMENTS (Continued)

TABLE 4.8.2.3-1

BATTERY CELL PARAMETERS REQUIREMENTS

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and ≤ 1/4 inch above maximum level indication mark (a)	>Minimum level indication mark, and ≤ 1/4 inch above maximum level indication mark (a)	Above top of plates, and not overflowing
Float Voltage	≥2.13 V	≥2.13 V	≥2.07 V
Specific Gravity (b) (c)	≥1.195	≥1.190 AND Average of all connected cells ≥1.200	Not more than 0.020 below average of all connected cells AND Average of all connected cells ≥1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charge provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 3 amps when on float charge.

(c) Or battery charging current is < 3 amps when on float charge. This is acceptable only during a maximum of 7 days following a battery recharge.

ELECTRICAL POWER SYSTEMS

125-VOLT D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

2 - 125-volt D.C. busses, and

2 - 125-volt batteries, each with at least one full capacity charger, associated with each of the above D.C. busses.

APPLICABILITY: MODES 5 and 6.
During movement of irradiated fuel assemblies.

ACTION:

With less than the above complement of D.C. equipment and busses OPERABLE, immediately declare the affected required features inoperable, or suspend all operations involving CORE ALTERATIONS, positive reactivity changes, and movement of irradiated fuel assemblies until the minimum required 125Volt D.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.8.2.4.1 The above required 125-volt D.C. busses shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt batteries and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

INSERT 1

4.8.2.5.1 Each D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and power availability.

4.8.2.5.2 Each 28-volt battery and above required charger shall be demonstrated OPERABLE:

INSERT 1

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8.2.5-1 meet Category A limits.

2. The overall battery voltage is greater than or equal to 27 volts on float charge.

INSERT 1

b. At least once per 92 days and once within 24 hours after a battery discharge < 25.7 V and once within 24 hours after a battery overcharge > 35 V by verifying that the parameters in Table 4.8.2.5-1 meet the Category B limits.

c. At least once per 92 days by verifying that:

INSERT 1

1. There is no visible corrosion at terminals or connectors or the connection resistance is:

≤ 50 micro ohms for inter-cell connections,
≤ 200 micro ohms for inter-tier connections,
≤ 70 micro ohms for field cable terminal connections, and
≤ 500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-tier connections (including cable resistance) and all field terminal connections at the battery.

INSERT 1

2. The average electrolyte temperature of the representative cells is ≥ 65°F.

d. At least once per 12 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

2. Remove visible terminal corrosion and verify cell-to-cell and terminal connections are coated with anti-corrosion material.

3. The connection resistance is:

≤ 50 micro ohms for inter-cell connections,
≤ 200 micro ohms for inter-tier connections,
≤ 70 micro ohms for field cable terminal connections, and
≤ 500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-tier connections (including cable resistance) and all field terminal connections at the battery.

SURVEILLANCE REQUIREMENTS (Continued)

INSERT 1 e. At least once per 18 months by verifying that the battery charger will supply ≥ 150 amperes at ≥ 28 volts for ≥ 4 hours.

INSERT 1 f. 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.

INSERT 1 g. 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. Satisfactory completion of this performance discharge test shall also satisfy the requirements of Specification 4.8.2.5.2.f if the performance discharge test is conducted during a shutdown where that test and the battery service test would both be required.

h. At least once per 12 months, during shutdown, if the battery shows signs of degradation OR has reached 85% of the service life with a capacity less than 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its capacity on the previous performance test, or is below 90% of the manufacturer's rating.

i. At least once per 24 months, during shutdown, if the battery has reached 85% of the service life with capacity greater than or equal to 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test.

[INFORMATION ONLY]

TABLE 4.8.2.5-1

BATTERY CELL PARAMETER REQUIREMENTS

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark and ≤ 1/4 inch above maximum level indication mark[a]	>Minimum level indication mark and ≤ 1/4 inch above maximum level indication mark[a]	Above top of plates and not overflowing
Float Voltage	≥2.13 V	≥2.13 V	≥2.07 V
Specific Gravity[b][c]	≥1.195	≥1.190 AND Average of all Connected cells ≥1.200	Not more than 0.020 below the average of all connected cells AND Average of all connected cells ≥1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charge provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) Or battery charging current is < 2 amps when on float charge. This is acceptable only during a maximum of 7 days following a battery recharge.

ELECTRICAL POWER SYSTEMS

28-VOLT D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.6 As a minimum, the following D. C. electrical equipment and bus shall be energized and OPERABLE:

1 - 28-volt D.C. bus, and

1 - 28-volt battery and at least one full capacity charger associated with the above D.C. bus.

APPLICABILITY: MODES 5 and 6.
During movement of irradiated fuel assemblies.

ACTION:

With less than the above complement of D.C. equipment and busses OPERABLE, immediately declare the affected required features inoperable, or suspend all operations involving CORE ALTERATIONS, positive reactivity changes, and movement or irradiated fuel assemblies until the minimum required 28Volt D.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.2.6.1 The above required 28-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the bus.

INSERT 1

4.8.2.6.2 The above required 28-volt batteries and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.5.2.

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.3.1 All containment penetration conductor overcurrent protective devices required to provide thermal protection of penetrations shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping either the primary or backup protective device, or racking out or removing the primary or backup device within 72 hours, declare the affected system or component inoperable, and verify the primary or backup protective device to be tripped, or the primary or backup device racked out or removed at least once per 7 days thereafter; or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 All required containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

INSERT 1

a. ~~At least once per 18 months~~:

1. For ~~at least one~~ 4.16 KV reactor coolant pump circuit(s), ~~such that all reactor coolant pump circuits are demonstrated OPERABLE at least once per 72 months~~, by performance of:

   (a) A CHANNEL CALIBRATION of the associated protective relays, and

   (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.

SURVEILLANCE REQUIREMENTS (Continued)

2. By verifying the OPERABILITY of the required molded case and lower voltage circuit breakers, by selecting and functionally testing a representative sample of at least 10% of all the circuit breakers of that type. Circuit breakers selected for functional testing shall be selected on a rotating basis. The functional test shall consist of injecting a current input at the specified setpoint to each selected circuit breaker and verifying that each circuit breaker functions as designed. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during the 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.

INSERT 1

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.

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of the Reactor Coolant System, the fuel storage pool, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 6 (Only applicable to the refueling canal, the fuel storage pool and refueling cavity when connected to the Reactor Coolant System)

ACTION:
With the requirements of the above specification not satisfied, immediately

a. Suspend CORE ALTERATIONS and
b. Suspend positive reactivity additions and
c. Initiate action to restore boron concentration to within limit specified in the COLR.
d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1 Verify the boron concentration is within the limit of the COLR every 72 hours

INSERT 1

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, 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, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the required monitors inoperable, determine the boron concentration of the reactor coolant system at least once per 12 hours.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK ~~at least once per 12 hours~~, and

b. A CHANNEL FUNCTIONAL TEST ~~at least once per 7 days~~.

INSERT 1

REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch inside door is capable of being closed and held in place by a minimum of four bolts, or an equivalent closure device installed and capable of being closed,

b. A minimum of one door in each airlock is capable of being closed

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
2. capable of being closed by the Containment Purge and Pressure-Vacuum Relief Isolation System.

Note: Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

INSERT 1

~~4.9.4.1~~ Each of the above required containment building penetrations shall be determined to be either in its required condition or capable of being closed by a manual or automatic containment isolation valve ~~at least once per 7 days~~.

4.9.4.2 Once per refueling prior to the start of movement of irradiated fuel assemblies within the containment building, verify the capability to close, within 1 hour, the equipment hatch inside door or an equivalent closure device. Applicable only when the equipment hatch is open during movement of irradiated fuel in the containment building.

INSERT 1

4.9.4.3 Verify, ~~once per 18 months~~, each required containment purge isolation valve actuates to the isolation position on a manual actuation signal.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

ALL WATER LEVELS

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop 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 pressure vessel hot legs.

c. The provisions of Specification 3.0.3 are not applicable.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.9.8.1 ~~At least once per 12 hours~~ one RHR loop shall be verified in operation and circulating coolant at a flow rate of:

a. greater than or equal to 1000 gpm, and

b. sufficient to maintain the RCS temperature at less than or equal to 140°F.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movements of fuel assemblies or control rods.

3/4.9.11 STORAGE POOL WATER LEVEL

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:

With the requirements of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore water level to within its limit within 4 hours. The provisions of Specification 3.0.3 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.

INSERT 1

3/4.9.12 FUEL HANDLING AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The Fuel Handling Area Ventilation System shall be OPERABLE with:

a. Two exhaust fans and one supply fan OPERABLE and operating, and

b. Capable of maintaining slightly negative pressure in the Fuel Handling Building.

APPLICABILITY: During movement of irradiated fuel within the Fuel Handling Building

ACTION:

a. With no Fuel Handling Area Ventilation System OPERABLE, suspend all operations involving movement of fuel within the storage pool until the Fuel Handling Area Ventilation System is restored to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required ventilation system shall be demonstrated OPERABLE:

INSERT 1

a. At least once per 12 hours by verifying that the Fuel Handling Building is maintained at a slightly negative pressure with respect to atmospheric pressure.

b. At least once per 31 days by verifying both exhaust fans and one supply fan start and operate for at least 15 minutes, if not operating already.

c. At least once per 18 months by verifying a system flowrate of 19,490 cfm ± 10% during system operation.

3/4.10 SPECIAL TEST EXCEPTIONS

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 the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or FULLY WITHDRAWN shall be determined ~~at least once per 2 hours~~ INSERT 1

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

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, 3.1.3.4, 3.1.3.5, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85%# of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 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.

INSERT 1

4.10.2.2 The below listed surveillance requirements shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Surveillances 4.2.2.2 and 4.2.2.3.

b. Surveillances 4.2.3.1 and 4.2.3.2.

INSERT 1

See page 3/4 10-3

SPECIAL TEST EXCEPTIONS

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, 3.1.3.4 and 3.1.3.5 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 531°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 531°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER ~~at least once per hour~~ during PHYSICS TESTS. INSERT 1

4.10.3.2 Each Intermediate and Power Range Channel shall ~~be subjected~~ to a CHANNEL FUNCTIONAL TEST prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 531°F ~~at least once per 30 minutes~~ during PHYSICS TESTS.

INSERT 1

SPECIAL TEST EXCEPTIONS

NO FLOW TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set less than or equal to 25% of RATED THERMAL POWER

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint ~~at least once per hour during startup and~~ PHYSICS TESTS.

INSERT 1

4.10.4.2 Each Intermediate, Power ~~Range Channel and~~ P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST prior to initiating startup or PHYSICS TESTS.

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outdoor 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 above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each outdoor temporary tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents ~~at least once per 7 days~~ when radioactive materials are being added to the tank.

INSERT 1

* Tanks included in this Specification are those outdoor temporary 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.

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume.

APPLICABILITY: At all times. *

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously** monitoring the waste gases in the waste gas holdup system with the oxygen monitor. If hydrogen is not measured, the concentration of hydrogen shall be assumed to exceed 4% by volume.

* Not applicable to portions of the Waste Gas System removed from service for maintenance provided that, the portions removed for maintenance are isolated, and purged of hydrogen to less than 4% by volume.

** If the oxygen monitoring instrumentation is inoperable, operation of the waste gas holdup system may continue provided grab samples are collected ~~at least once per 24 hours~~ and analyzed within the following 4 hours.

INSERT 1

4. When the W* methodology has been implemented, inspect 100 percent of the inservice tubes for the entire hot-leg tubesheet W* distance with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 6.8.4.i.c.1 every 24 effective full power months or one refueling outage (whichever is less).

e. Provisions for monitoring operational primary-to-secondary leakage.

INSERT 3

ATTACHMENT 4

License Amendment Request

Salem Nuclear Generating Station - Units 1 and 2
NRC Docket Nos. 50-272 and 50-311

Proposed Changes to Technical Specification Bases Pages

Unit 1 TS Bases Pages

B 3/4 1-5
B 3/4 2-6
B 3/4 3-1
B 3/4 3-1a
B 3/4 3-2
B 3/4 3-4
B 3/4 3-5
B 3/4 4-2
B 3/4 4-4b
B 3/4 4-17
B 3/4 6-1d
B 3/4 7-5e
B 3/4 7-11
B 3/4 8-3
B 3/4 8-4
B 3/4 9-1b

Unit 2 TS Bases Pages

B 3/4 1-5
B 3/4 2-6
B 3/4 3-1
B 3/4 3-1a
B 3/4 3-2
B 3/4 3-4
B 3/4 3-5
B 3/4 4-2
B 3/4 4-4a
B 3/4 4-5
B 3/4 4-18
B 3/4 6-1d
B 3/4 7-5e
B 3/4 7-11
B 3/4 8-3
B 3/4 8-4
B 3/4 9-1b

INSERT 1

In accordance with the Surveillance Frequency Control Program

INSERT 2

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The ACTION statements which permit limited variation from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Mis-alignment 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. The reactivity worth of a mis-aligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumption used in the accident analysis.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg} >541°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

INSERT 1

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

INSERT 2

The terms "Shutdown Rod Position Indicator", "Analog Rod Position Indicator", "Control Rod Position Indicator", and Rod Position Indicator" are all used in this bases section or in the Technical Specifications, and all refer to indication driven by the output of the Analog Rod Position Indication (ARPI) system.

One method for determining rod position are the indicators on the control console. An alternate method of determining rod position is the plant computer. Either the control console indicator or plant computer is sufficient to comply with this specification. The plant computer receives the same input from ARPI as the control console indicators and provides resolution equivalent to or better than the control console indicators. The plant computer also provides a digital readout of rod position which eliminates interpolation and parallax errors inherent to analog scales.

Rod demand position is indicated on the control console and the plant computer. The rod demand position is a digital signal, namely a pulse, and is generated each time the Rod Control System demands a rod position step change, one pulse for each rod step. The pulses are "counted" and displayed by the control console group demand step counters. There are two group demand step counters for each bank of rods with exception of shutdown banks C and D. The plant computer also "counts" and displays the demand pulses. Only the group 1 demand position of each rod bank is displayed on the plant computer as only the group 1 pulses are routed to the plant computer. The group 1 demand position on the plant computer is, by default, called "Cont Bank A Steps" or "S/D Bank A Steps" etc. with no reference to group 1 or group 2.

As the plant computer receives the same demand pulses from the Rod Control System as the control console group demand step counters and provides equivalent resolution, the plant computer "bank step" display provides an alternate method of determining group 1 rod demand position. Either the control console group 1 demand step counter or the plant computer "bank step" display is sufficient to comply with this specification for group 1 rod demand position. Only the control console group 2 demand counter can be used to comply with the specification for group 2 rod demand.

 Revised by letter dated 9/12/01

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 the design DNBR value throughout each analyzed transient.

~~The 12 hour periodic surveillance of these parameters thru 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 flow rate on a 12 hour basis.~~

INSERT 2

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as-left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration + comparator setting accuracy).

The Trip Setpoints used in the bistables are based on the analytical limits stated in the UFSAR. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those Reactor Protection System (RPS) channels that must function in harsh environments as defined by 10 CFR 50.49, the Trip Setpoints and Allowable Values specified in the Technical Specification Limiting Conditions for Operation (LCO's) are conservatively adjusted with respect to the analytical limits. The methodology used to calculate the Trip Setpoints is consistent with Instrument Society of America standard ISA-S67.04-1982, which is endorsed via NRC Regulatory Guide 1.105, Rev. 2. The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the safety analyses which demonstrate that safety limits are not violated remain valid (provided the unit is operated within the LCO's at the onset of any design basis event and the equipment functions as designed).

The Trip Setpoints and Allowable Values listed in the LCO's incorporate all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

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

INSERT 2

Instrumentation System," and Supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

INSERT 2

The verification 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 analysis. No credit is taken in the analysis for those channels with response times indicated as not applicable (i.e., N.A.). The FSAR tables 7.3-8 Note 8 response times for feedwater isolation are based on WCAP-16503, "Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit (CFCU) Margin Recovery Project," Revision 3, (LCR S06-10). SGFP trip and FIV closure are credited in the containment analyses for LOCA and MSLB in case an FRV fails open.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocation for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

In the postulated Fuel Handling Accident, the revised dose calculations, performed using 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term, do not take credit for automatic containment purge isolation thus allowing for continuous monitoring of containment activity until containment closure is achieved. If required, containment purge isolation can be initiated manually from the control room.

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION (Continued)

CROSS REFERENCE - TABLES 3.3-6 AND 4.3-3

T/S Table Item No.	Instrument Description	Acceptable RMs Channels
1a	Fuel Storage Area	1R5 or 1R9
1b	DELETED	
2a1a	Containment Gaseous Activity Purge & Pressure/Vacuum Relief Isolation	1R12A or 1R41A, and D(1) (2)
2a1b	Containment Gaseous Activity RCS Leakage Detection	1R12A
2a2a	(NOT USED)	
2a2b	Containment Air Particulate Activity RCS Leakage Detection	1R11A
2b1	Noble Gas Effluent Medium Range Auxiliary Building Exhaust System (Plant Vent)	1R41B & D (1) (3) (5)
2b2	Noble Gas Effluent High Range Auxiliary Building Exhaust System (Plant Vent)	1R41C & D (1) (4) (5)
2b3	Noble Gas Effluent Condenser Exhaust System	1R15
3a	Unit 1 Control Room Intake Channel 1 (to Unit 1 Monitor)	1R1B-1
	Unit 1 Control Room Intake Channel 2 (to Unit 2 Monitor)	2R1B-2
	Unit 2 Control Room Intake Channel 1 (to Unit 2 Monitor)	2R1B-1
	Unit 2 Control Room Intake Channel 2 (to Unit 1 Monitor)	1R1B-2

Immediate action(s), in accordance with the LCO Action Statements, means that the required action should be pursued without delay and in a controlled manner.

(1) The channels listed are required to be operable to meet a single operable channel for the Technical Specification's "Minimum Channels Operable" requirement.

(2) The setpoint applies to 1R41D. The measurement range applies to 1R41A and B which display in uCi/cc using the appropriate channel conversion factor from cpm to uCi/cc.

INSERT 1

3/4.3.3.14 POWER DISTRIBUTION MONITORING SYSTEM (PDMS)

The Power Distribution Monitoring System (PDMS) provides core monitoring of the limiting parameters. The PDMS continuous core power distribution measurement methodology begins with the periodic generation of a highly accurate 3-D nodal simulation of the current reactor power distribution. The simulated reactor power distribution is then continuously adjusted by nodal and thermocouple calibration factors derived from an incore power distribution measurement obtained using the incore movable detectors to produce a highly accurate power distribution measurement. The nodal calibration factors are updated at least once every 180 Effective Full Power Days (EFPD). Between calibrations, the fidelity of the measured power distribution is maintained via adjustment to the calibrated power distribution provided by continuously input plant and core condition information. The plant and core condition data utilized by the PDMS is cross checked using redundant information to provide a robust basis for continued operation. The loop inlet temperature is generated by averaging the respective temperatures from each of the loops, excluding any bad data. The core exit thermocouples provide many readings across the core and by the nature of their usage with the PDMS, smoothing of the measured data and elimination of bad data is performed with the Surface Spline fit. PDMS uses the NIS Power Range excore detectors to provide information on the axial power distribution. Hence, the PDMS averages the data from the four Power Range excore detectors and eliminates any bad excore detector data.

The bases for the operability requirements of the PDMS is to provide assurance of the accuracy and reliability of the core parameters measured and calculated by the PDMS core power distribution monitor function. These requirements fall under four categories:

1. Assure an adequate number of operable critical sensors.
2. Assure sufficiently accurate calibration of these sensors.
3. Assure an adequate calibration database regarding the number of data sets.
4. Assure the overall accuracy of the calibration.

The minimum number of required plant and core condition inputs include the following:

1. Control Bank Positions.
2. At least 50% of the cold leg temperatures.
3. At least 75% of the signals from the power range excore detector channels (comprised of top and bottom detector section).
4. Reactor Power Level.
5. A minimum number and distribution of operable core exit thermocouples.
6. A minimum number and distribution of measured fuel assembly power distribution information obtained using the incore movable detectors is incorporated in the nodal model calibration information.

The sensor calibration of Items 1, 2, 3, and 4 above are covered under other specifications. Calibration of the core exit thermocouples is accomplished in two parts. The first being a sensor specific correction to K-type thermocouple temperature indications based on data from a cross calibration of the thermocouple temperature indications to the average RCS temperature measured via the RTDs under isothermal RCS conditions. The second part of the thermocouple calibration is the generation of thermocouple flow mixing

factors that cause the radial power distribution measured via the thermocouples to agree with the radial power distribution from a full core flux map measured using the incore movable detectors. This calibration is updated at least once every 180 EFPD.

The operability requirements previously contained in Specification 3.3.3.2 have been moved to UFSAR Section 7.7.2.8 as part of Amendment 282.

INSERT 1

3/4.3.4 DELETED

3/4.4.4 PRESSURIZER

INSERT 2

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to assure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE assures that the plant will be able to establish natural circulation.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. For Refueling Outage 18 and the subsequent operating cycle only, the following definition applies: A SG tube is defined as the length of the tube beginning 17 inches from the top of the tubesheet on the tube hot leg side to 17 inches below the top of the tubesheet on the tube cold leg side as defined in LCR S07-01 (including WCAP-16640-P). The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.i, "Steam Generator (SG) Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that significantly affect burst or collapse. In that context, the term "significant" is defined as, "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established."

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

its potential consequences. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. The reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary. The action times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Surveillances

Verifying RCS leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. Unidentified leakage and identified leakage are determined by performance of an RCS water inventory balance. The RCS water inventory must be met with the reactor at steady state conditions. The surveillance is modified by a Note that the surveillance is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The 72 hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

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Mode ascension to MODE 1-3 is acceptable without a current RCS Inventory Balance, provided the asterisked note of "Not required to be completed until 12 hours after establishment of steady state operations", is complied with.

Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If SR 4.4.6.2.c is not met, compliance with LCO 3.4.5, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines). If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one Steam Generator. The Surveillance is modified by a Note which states that the surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary-to-secondary leakage determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The Surveillance Frequency of 72 hours is a reasonable interval to trend primary-to-secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines).

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3/4.4.7

THIS SECTION DELETED

3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.11 THIS SECTION INTENTIONALLY BLANK

3/4.4.12 REACTOR VESSEL HEAD VENTS

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

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertant or irreversible actuation while ensuring that a single failure in a vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System Vent Systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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Correction letter dated February 15, 1990, to Amendment 108 dated January 29, 1990.

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when the containment air lock door is opened, this test is only required to be performed upon entering or exiting a containment air lock but is not required more frequently than every six months. The six-month frequency is based on engineering judgement and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

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In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREACS train cannot be restored to OPERABLE status, align CREACS for single filtration train operation within 4 hours, or suspend movement of irradiated fuel assemblies. With CREACS aligned for single filtration train operation with one of the two remaining fans or associated outlet damper inoperable, restore the fan or damper to OPERABLE status within 72 hours. The 72 hours completion time is based on the ability of the remaining train components to provide the required capability.

In MODE 5 or 6, or during the movement of irradiated fuel assemblies, with two CREACS trains inoperable or with one or more CREACS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

Immediate action(s), in accordance with the LCO Action Statements, means that the required action should be pursued without delay and in a controlled manner.

SURVEILLANCE REQUIREMENTS

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. The 31-day Frequency is based on the reliability of the equipment and the two-train redundancy. TS Surveillance Requirement verifies that each fan is capable of operating for at least 15 minutes by initiating flow through the HEPA filter and charcoal adsorbers train(s) to ensure that the system is available in a standby mode.

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Filter testing verifies that the required CREACS testing is performed in accordance with the surveillance requirements. The surveillance requirements include testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the surveillance requirements. Filter testing will be in accordance with the applicable sections of ANSI N510 (1975) with the exception that laboratory testing of activated carbon will be in accordance with ASTM D3803 (1989). The acceptance criteria for the laboratory testing of the carbon adsorber is determined by applying a minimum safety factor of 2 to the charcoal adsorber removal efficiency credited in the design basis dose analysis as specified in Generic Letter 99-02.

Actuation testing verifies that each CREACS train starts and operates on an actual or simulated actuation signal. The Frequency of 18-months is based on industry operating experience and is consistent with the typical refueling cycle.

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3/4.7.11 FUEL STORAGE POOL BORON CONCENTRATION (continued)

The Required Actions are modified indicating that LCO 3.0.3 does not apply. Storage of fuel assemblies and the boron concentration in the spent fuel storage pool are independent of reactor operation. Therefore TS 3/4 3.7.11 and TS 3/4 3.7.12 include the exception to LCO 3.0.3 to preclude an inappropriate reactor shutdown. When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving fuel assemblies in the spent fuel pool while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving fuel assemblies in spent fuel pool while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

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not available. Therefore, it is required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required shutdown margin or boron concentration. Suspending positive reactivity additions that could result in failure to meet the minimum shutdown margin or boron concentration limit is required to assure continued safe operation.

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The Surveillance Requirements for ~~demonstrating the OPERABILITY~~ of the diesel generators are based upon the recommendations of Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977. Regulatory Guide 1.108 criteria for determining and reporting valid tests and failures, and accelerated diesel generator testing, have been superseded by implementation of the Maintenance Rule for the diesel generators per 10CFR50.65. In addition to the Surveillance Requirements of 4.8.1.1.2, diesel preventative maintenance is performed in accordance with procedures based on manufacturer's recommendations with consideration given to operating experience.

The minimum voltage and frequency stated in the Surveillance Requirements (SR) are those necessary to ensure the Emergency Diesel Generator (EDG) can accept Design Basis Accident (DBA) loading while maintaining acceptable voltage and frequency levels. Stable operation at the nominal voltage and frequency values is also essential in establishing EDG OPERATILITY, but a time constraint is not imposed. The lack of a time constraint is based on the fact that a typical EDG will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not dampened out by load application. In lieu of a time constraint in the SR, controls will be provided to monitor and trend the actual time to reach stable operation within the band as a means of ensuring there is no voltage regulator or governor degradation that could cause an EDG to become inoperable.

"Standby condition" for the purpose of defining the condition of the engine immediately prior to starting for surveillance requirements requires that the lube oil temperature be between 100 °F and 170 °F. The minimum lube oil temperature for an OPERABLE diesel is 100 °F.

The thirteen second time requirement for the Emergency Diesel Generator to reach rated voltage and frequency was originally based on a Westinghouse assumption of fifteen seconds that included the delay time between the occurrence of the incident and the application of electrical power to the first sequenced safeguards pump (BURL-3011, dated November 13, 1974) and included an instrument response time of two seconds (BURL-1531, dated July 27, 1970). The times specified in UFSAR Section 15.4 bound the thirteen seconds specified in the TS.

The narrower band for frequency specified for testing performed in steady state isochronous operation will ensure the EDG will not be run in an overloaded condition (steady state) during accident conditions. Steady state is assumed to be achieved after one minute of operation in the isochronous mode with all required loads sequenced on the bus.

The narrower band for steady state voltage is specified for operation when

the EDG is not synchronized to the grid to ensure the voltage regulator will protect driven equipment from over-voltages during accident conditions. Procedural controls will ensure that equipment voltages are maintained within acceptable limits during testing when paralleled to the grid.

The wider band for frequency is appropriate for testing done with the governor in the droop mode. Likewise the wider band for voltage is appropriate when paralleled to the grid.

All voltages and frequencies specified in SR 4.8.1.1.2 are representative of the analytical values and do not account for postulated instrument inaccuracy. Instrument inaccuracies for EDG voltage and frequency are administratively controlled.

Preventive maintenance includes those activities (including pro-test inspections, measurements, adjustments and preparations) performed to maintain an otherwise OPERABLE EDG in an OPERABLE status. Corrective maintenance includes those activities required to correct a condition that would cause the EDG to be inoperable.

Surveillance requirement 4.8.1.2 is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of the surveillance requirement, and to preclude de-energizing a required ESF bus or disconnecting a required offsite circuit during performance of surveillance requirements. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these surveillance requirements must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit are required to be OPERABLE. During Startup, prior to entering Mode 4, the surveillance requirements are required to be completed if the surveillance frequency has been exceeded or will be exceeded prior to the next scheduled shutdown.

3/4.8.3 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.

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The surveillance frequency applicable to molded case circuit breakers and lower voltage circuit breakers provides assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of molded case and lower voltage circuit breakers. Each manufacturer's molded case circuit breakers and lower voltage 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 molded case or lower voltage circuit breakers, it is necessary to further divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

Containment penetration conductor overcurrent protective device information is provided in the UFSAR.

In addition to immediately suspending CORE ALTERATIONS and positive reactivity additions, boration to restore the concentration must be initiated immediately. In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions. Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

The Surveillance Requirement (SR) ensures that the coolant boron concentration in the RCS, and connected portions of the refueling canal, the fuel storage pool and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to reconnecting portions of the refueling canal, the fuel storage pool or the refueling cavity to the RCS, this SR must be met per SR 4.0.4. If any dilution activity has occurred while the cavity or canal was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS. A minimum frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The frequency is based on operating experience, which has shown 72 hours to be adequate.

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3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. The 80-hour decay time (LAR S08-01) is consistent with the assumptions used in the fuel handling accident analyses and the resulting dose calculations using the Alternative Source Term described in Reg. Guide 1.183.

The minimum requirement for reactor subcriticality also ensures that the decay time is consistent with that assumed in the Spent Fuel Pool cooling analysis. The calendar based restrictions are established for the actual movement of irradiated fuel; i.e., movement cannot commence in the October 15th through May 15th window unless at least 80 hours has elapsed since subcriticality was achieved. The 80 hour clock can start prior to October 15 but must end in the October 15th - May 15th window for the 80 hour criteria to be applicable.

The ACTION statements which permit limited variation from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Mis-alignment 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. The reactivity worth of a mis-aligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumption used in the accident analysis.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg} >541°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

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Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

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The terms "Shutdown Rod Position Indicator," "Analog Rod Position Indicator," "Control Rod Position Indicator," and "Rod Position Indicator" are all used in this bases section or in Technical Specifications, and all refer to indication driven by the output of the Analog Rod Position Indication (ARPI) system.

One method for determining rod position are the indicators on the control console. An alternate method of determining rod position is the plant computer. Either the control console indicator or plant computer is sufficient to comply with this specification. The plant computer receives the same input from ARPI as the control console indicators and provides resolution equivalent to or better than the control console indicators. The plant computer also provides a digital readout of rod position which eliminates interpolation and parallax errors inherent to analog scales.

Rod demand position is indicated on the control console and the plant computer. The rod demand position is a digital signal, namely a pulse, and is generated each time the Rod Control System demands a rod position step change, one pulse for each rod step. The pulses are "counted" and displayed by the control console group demand step counters. There are two group demand step counters for each bank of rods with exception of shutdown banks C and D. The plant computer also "counts" and displays the demand pulses. Only the group 1 demand position of each rod bank is displayed on the plant computer as only the group 1 pulses are routed to the plant computer. The group 1 demand position on the plant computer is, by default, called "Cont Bank A Steps" or "S/D Bank A Steps" etc. with no reference to group 1 or group 2.

As the plant computer receives the same demand pulses from the Rod Control System as the control console group demand step counters and provides equivalent resolution, the plant computer "bank step" display provides an alternate method of determining group 1 rod demand position. Either the control console group 1 demand step counter or the plant computer "bank step" display is sufficient to comply with this specification for group 1 rod demand position. Only the control console group 2 demand counter can be used to comply with the specification for group 2 rod demand.

 Revised by letter dated 9/12/01

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2 hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3% from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained with 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 the design DNBR value throughout each analyzed transient.

~~The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. 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 flow rate on a 12 hour basis.~~

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3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as-left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration + comparator setting accuracy).

The Trip Setpoints used in the bistables are based on the analytical limits stated in the UFSAR. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those Reactor Protection System (RPS) channels that must function in harsh environments as defined by 10 CFR 50.49, the Trip Setpoints and Allowable Values specified in the Technical Specification Limiting Conditions for Operation (LCO's) are conservatively adjusted with respect to the analytical limits. The methodology used to calculate the Trip Setpoints is consistent with Instrument Society of America standard ISA-S67.04-1982, which is endorsed via NRC Regulatory Guide 1.105, Rev. 2. The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the safety analyses which demonstrate that safety limits are not violated remain valid (provided the unit is operated within the LCO's at the onset of any design basis event and the equipment functions as designed).

The Trip Setpoints and Allowable Values listed in the LCO's incorporate all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

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

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Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and Supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The verification 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 analysis. No credit is taken in the analysis for those channels with response times indicated as not applicable (i.e., N.A.). The Note 8 response times for feedwater isolation are based on WCAP-16503, "Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit (CFCU) Margin Recovery Project," Revision 3, (LCR S06-10). SGFP trip and FIV closure are credited in the containment analyses for LOCA and MSLB in case an FRV fails open.

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Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocation for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

In the postulated Fuel Handling Accident, the revised dose calculations, performed using 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term, do not take credit for automatic containment purge isolation thus allowing for continuous monitoring of containment activity until containment closure is achieved. If required, containment purge isolation can be initiated manually from the control room.

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION (Continued)

CROSS REFERENCE - TABLES 3.3-6 and 4.3-3

T/S Table Item No.	Instrument Description	Acceptable RMs Channels
1a	Fuel Storage Area	2R5 or 2R9
1b	DELETED	
2a1a	Containment Gaseous Activity Purge & Pressure/Vacuum Relief Isolation	2R12A or 2R41A, B and D(1) (2)
2a1b	Containment Gaseous Activity RCS Leakage Detection	2R12A
2a2a	(NOT USED)	
2a2b	Containment Air Particulate Activity RCS Leakage Detection	2R11A
2b1	Noble Gas Effluent Medium Range Auxiliary Building Exhaust System (Plant Vent)	2R45B(3)
2b2	Noble Gas Effluent High Range Auxiliary Building Exhaust System (Plant Vent)	2R45C(3)
2b3	Noble Gas Effluent Condenser Exhaust System	2R15
3a	Unit 2 Control Room Intake Channel 1 (to Unit 2 Monitor) Unit 2 Control Room Intake Channel 2 (to Unit 1 Monitor) Unit 1 Control Room Intake Channel 1 (to Unit 1 Monitor) Unit 1 Control Room Intake Channel 2 (to Unit 2 Monitor)	2R1B-1 1R1B-2 1R1B-1 2R1B-2

(1) The channels listed are required to be operable to meet a single operable channel for the Technical Specification's "Minimum Channels Operable" requirement.

(2) For Modes 1, 2, 3, 4 & 5, the setpoint applies to 2R41D per Specification 3.3.3.9. The measurement range applies to 2R41A and B which display in uCi/cc using the appropriate channel conversion factor from cpm to uCi/cc.

(3) If 2R45 is out of service 2R41 may be used to meet the technical specification action requirement.

3/4.3.4 Deleted

3/4.3.3.14 POWER DISTRIBUTION MONFTORING SYSTEM (PDMS)

The Power Distribution Monitoring System (PDMS) provides core monitoring of the limiting parameters. The PDMS continuous core power distribution measurement methodology begins with the periodic generation of a highly accurate 3-D nodal simulation of the current reactor power distribution. The simulated reactor power distribution is then continuously adjusted by nodal and thermocouple calibration factors derived from an incore power distribution measurement obtained using the incore movable detectors to produce a highly accurate power distribution measurement. The nodal calibration factors are updated at least once every 180 Effective Full Power Days (EFPD). Between calibrations, the fidelity of the measured power distribution is maintained via adjustment to the calibrated power distribution provided by continuously input plant and core condition information. The plant and core condition data utilized by the PDMS is cross checked using redundant information to provide a robust basis for continued operation. The loop inlet temperature is generated by averaging the respective temperatures from each of the loops, excluding any bad data. The core exit thermocouples provide many readings across the core and by the nature of their usage with the PDMS, smoothing of the measured data and elimination of bad data is performed with the Surface Spline fit. PDMS uses the NIS Power Range excore detectors to provide information on the axial power distribution. Hence, the PDMS averages the data from the four Power Range excore detectors and eliminates any bad excore detector data.

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The bases for the operability requirements of the PDMS is to provide assurance of the accuracy and reliability of the core parameters measured and calculated by the PDMS core power distribution monitor function. These requirements fall under four categories:

1. Assure an adequate number of operable critical sensors.

2. Assure sufficiently accurate calibration of these sensors.

3. Assure an adequate calibration database regarding the number of data sets.

4. Assure the overall accuracy of the calibration.

The minimum number of required plant and core condition inputs include the following:

1. Control Bank Positions.

2. At least 50% of the cold leg temperatures.

3. At least 75% of the signals from the power range excore detector channels (comprised of top and bottom detector section).

4. Reactor Power Level.

5. A minimum number and distribution of operable core exit thermocouples.

6. A minimum number and distribution of measured fuel assembly power distribution information obtained using the incore movable detectors is incorporated in the nodal model calibration information.

The sensor calibration of Items 1, 2, 3, and 4 above are covered under other specifications. Calibration of the core exit thermocouples is accomplished in two parts. The first being a sensor specific correction to K-type thermocouple temperature indications based on data from a cross calibration of the thermocouple temperature indications to the average RCS temperature measured via the RTDs under isothermal RCS conditions. The second part of the thermocouple calibration is the generation of thermocouple flow mixing factors that cause the radial power distribution measured via the thermocouples to agree with the radial power distribution from a full core flux map measured using the incore movable detectors. This calibration is updated at least once every 180 EFPD.

The operability requirements previously contained in Specification 3.3.3.2 have been moved to UFSAR Section 7.7.2.8 as part of Amendment 265.

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3/4.4.2 and 3/4.4.3 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 pounds 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 loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperature. While in Mode 5 the safety valve requirement may be met by establishing a vent path of equivalent relieving capacity when no code safety valves are OPERABLE.

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 Protective 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.

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.

Surveillance testing allows a ± 3% lift setpoint tolerance. However, to allow for drift during subsequent operation, the valves must be reset to within ± 1% of the lift setpoint following testing.

3/4.4.4 PRESSURIZER

INSERT 2

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control RCS pressure and establish natural circulation.

3/4.4.5 RELIEF VALVES

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown.

3/4.4.7.2 OPERATIONAL LEAKAGE (Continued)

Actions

Unidentified leakage or identified leakage in excess of the LCO limits must be reduced to within limits within 4 hours. This action time allows time to verify leakage rates and either identify unidentified leakage or reduce leakage to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the reactor coolant pressure boundary (RCPB). If any pressure boundary leakage exists, or primary-to-secondary leakage is not within limit, or if unidentified or identified leakage cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. The reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary. The action times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Surveillances

Verifying RCS leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. Unidentified leakage and identified leakage are determined by performance of an RCS water inventory balance. The RCS water inventory must be met with the reactor at steady state operating conditions. The surveillance is modified by a Note that the surveillance is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. ~~The 72 hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.~~ INSERT 2

Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If SR 4.4.7.2.1.c is not met, compliance with LCO 3.4.6, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines). If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one Steam Generator. The Surveillance is modified by a Note that states that the surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary-to-secondary leakage determination, steady

3/4.4.7.2 OPERATIONAL LEAKAGE (Continued)

state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The Surveillance Frequency of 72 hours is a reasonable interval to trend primary-to-secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines).

INSERT 2

3/4.4.8

THIS SECTION DELETED

3/4.4.11 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 through 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 Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.12 REACTOR VESSEL HEAD VENTS

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

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertant or irreversible actuation while ensuring that a single failure vent in a valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System Vent Systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

INSERT 2

Correction letter dated February 15, 1990, to Amendment 86 dated January 29, 1990.

Maintaining containment airlocks OPERABLE requires compliance with the leakage rate test requirements of 10CFR50, Appendix J, as modified by approved exemptions. This Surveillance Requirement reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The frequency is required by Appendix J, as modified by approved exemptions. Thus, the provision of Specification 4.0.2 (which allows frequency extensions) does not apply.

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. ~~Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when the containment air lock door is opened, this test is only required to be performed upon entering or exiting a containment air lock but is not required more frequently than every six months. The six-month frequency is based on engineering judgement and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.~~

INSERT 2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREACS train cannot be restored to OPERABLE status, align CREACS for single filtration train operation within 4 hours, or suspend movement of irradiated fuel assemblies. With CREACS aligned for single filtration train operation with one of the two remaining fans or associated outlet damper inoperable, restore the fan or damper to OPERABLE status within 72 hours. The 72 hours completion time is based on the ability of the remaining train components to provide the required capability.

In MODE 5 or 6, or during the movement of irradiated fuel assemblies, with two CREACS trains inoperable or with one or more CREACS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

Immediate action(s), in accordance with the LCO Action Statements, means that the required action should be pursued without delay and in a controlled manner.

SURVEILLANCE REQUIREMENTS

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. The 31-day Frequency is based on the reliability of the equipment and the two-train redundancy. TS Surveillance Requirement verifies that each fan is capable of operating for at least 15 minutes by initiating flow through the HEPA filter and charcoal adsorbers train(s) to ensure that the system is available in a standby mode.

INSERT 2

Filter testing verifies that the required CREACS testing is performed in accordance with the surveillance requirements. The surveillance requirements include testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the surveillance requirements. Filter testing will be in accordance with the applicable sections of ANSI N510 (1975) with the exception that laboratory testing of activated carbon will be in accordance with ASTM D3803 (1989). The acceptance criteria for the laboratory testing of the carbon adsorber is determined by applying a minimum safety factor of 2 to the charcoal adsorber removal efficiency credited in the design basis dose analysis as specified in Generic Letter 99-02.

Actuation testing verifies that each CREACS train starts and operates on an actual or simulated actuation signal. The Frequency of 18-months is based on industry operating experience and is consistent with the typical refueling cycle.

INSERT 2

3/4.7.11 FUEL STORAGE POOL BORON CONCENTRATION (continued)

The Required Actions are modified indicating that LCO 3.0.3 does not apply. Storage of fuel assemblies and the boron concentration in the spent fuel storage pool are independent of reactor operation. Therefore TS 3/4 3.7.11 and TS3/ 4 3.7.12 include the exception to LCO 3.0.3 to preclude an inappropriate reactor shutdown. When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the fuel storage pool fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving fuel assemblies in the spent fuel pool while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving fuel assemblies in the spent fuel pool while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

INSERT 2

INSERT 2

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are based upon the recommendations of Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977. Regulatory Guide 1.108 criteria for determining and reporting valid tests and failures, and accelerated diesel generator testing, have been superseded by implementation of the Maintenance Rule for the diesel generators per 10CFR50.65. In addition to the Surveillance Requirements of 4.8.1.1.2, diesel preventative maintenance is performed in accordance with procedures based on manufacturer's recommendations with consideration given to operating experience.

The minimum voltage and frequency stated in the Surveillance Requirements (SR) are those necessary to ensure the Emergency Diesel Generator (EDG) can accept Design Basis Accident (DBA) loading while maintaining acceptable voltage and frequency levels. Stable operation at the nominal voltage and frequency values is also essential in establishing EDG OPERATILITY, but a time constraint is not imposed. The lack of a time constraint is based on the fact that a typical EDG will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not dampened out by load application. In lieu of a time constraint in the SR, controls will be provided to monitor and trend the actual time to reach stable operation within the band as a means of ensuring there is no voltage regulator or governor degradation that could cause an EDG to become inoperable.

"Standby condition" for the purpose of defining the condition of the engine immediately prior to starting for surveillance requirements requires that the lube oil temperature be between 100 °F and 170 °F. The minimum lube oil temperature for an OPERABLE diesel is 100 °F.

The thirteen second time requirement for the Emergency Diesel Generator to reach rated voltage and frequency was originally based on a Westinghouse assumption of fifteen seconds that included the delay time between the occurrence of the incident and the application of electrical power to the first sequenced safeguards pump (BURL-3011, dated November 13, 1974) and included an instrument response time of two seconds (BURL-1531, dated July 27, 1970). The times specified in UFSAR Section 15.4 bound the thirteen seconds specified in the TS.

The narrower band for frequency specified for testing performed in steady state isochronous operation will ensure the EDG will not be run in an overloaded condition (steady state) during accident conditions. Steady state is assumed to be achieved after one minute of operation in the isochronous mode with all required loads sequenced on the bus.

The narrower band for steady state voltage is specified for operation when the EDG is not synchronized to the grid to ensure the voltage regulator will protect driven equipment from over-voltages during accident conditions. Procedural controls will ensure that equipment voltages are maintained within acceptable limits during testing when paralleled to the grid.

The wider band for frequency is appropriate for testing done with the governor in the droop mode. Likewise the wider band for voltage is appropriate when paralleled to the grid.

All voltages and frequencies specified in SR 4.8.1.1.2 are representative of the analytical values and do not account for postulated instrument inaccuracy. Instrument inaccuracies for EDG voltage and frequency are administratively controlled.

Preventive maintenance includes those activities (including pro-test inspections, measurements, adjustments and preparations) performed to maintain an otherwise OPERABLE EDG in an OPERABLE status. Corrective maintenance includes those activities required to correct a condition that would cause the EDG to be inoperable.

Surveillance requirement 4.8.1.2 is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of the surveillance requirement, and to preclude de-energizing a required ESF bus or disconnecting a required offsite circuit during performance of surveillance requirements. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these surveillance requirements must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit are required to be OPERABLE. During Startup, prior to entering Mode 4, the surveillance requirements are required to be completed if the surveillance frequency has been exceeded or will be exceeded prior to the next scheduled shutdown.

3/4.8.3 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.

INSERT 2

~~The surveillance frequency applicable to molded case circuit breakers and lower voltage circuit breakers provides assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of molded case and lower voltage circuit breakers.~~ Each manufacturer's molded case circuit breakers and lower voltage 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 molded case or lower voltage circuit breakers, it is necessary to further divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

Containment penetration conductor overcurrent protective device information is provided in the UFSAR.

In addition to immediately suspending CORE ALTERATIONS and positive reactivity additions, boration to restore the concentration must be initiated immediately. In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions. Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

The Surveillance Requirement (SR) ensures that the coolant boron concentration in the RCS, and connected portions of the refueling canal, the fuel storage pool and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to reconnecting portions of the refueling canal, the fuel storage pool or the refueling cavity to the RCS, this SR must be met per SR 4.0.4. If any dilution activity has occurred while the cavity or canal was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS. A minimum frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The frequency is based on operating experience, which has shown 72 hours to be adequate.

INSERT 2

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. The 80-hour decay time (LAR S08-01) is consistent with the assumptions used in the fuel handling accident analyses and the resulting dose calculations using the Alternative Source Term described in Reg. Guide 1.183.

The minimum requirement for reactor subcriticality also ensures that the decay time is consistent with that assumed in the Spent Fuel Pool cooling analysis. The calendar based restrictions are established for the actual movement of irradiated fuel; i.e., movement cannot commence in the October 15th through May 15th window unless at least 80 hours has elapsed since subcriticality was achieved. The 80 hour clock can start prior to October 15 but must end in the October 15th - May 15th window for the 80 hour criteria to be applicable. Similarly, fuel movement between May 16th and October 14th cannot commence unless at least 168 hours has elapsed since subcriticality was achieved.

ATTACHMENT 5

License Amendment Request

Salem Nuclear Generating Station - Units 1 and 2
NRC Docket Nos. 50-272 and 50-311

Proposed No Significant Hazards Consideration

Description of Amendment Request: The change requests the adoption of an approved change to the standard technical specifications (STS) for Westinghouse Plants (NUREG-1431), to allow relocation of specific TS surveillance frequencies to a licensee-controlled program. The proposed change is described in Technical Specifications Task Force (TSTF) Traveler, TSTF-425, Revision 3 (Rev. 3) (ADAMS Accession No. ML090850642) related to the Relocation of Surveillance Frequencies to Licensee Control-RITSTF Initiative 5b and was described in the Notice of Availability published in the Federal Register on July 6, 2009 (74 FR 31996).

The proposed changes are consistent with NRC-approved Industry/Technical Specifications Task Force (TSTF) Traveler, TSTF-425, Rev. 3, "Relocate Surveillance Frequencies to Licensee Control-RITSTF Initiative 5b." The proposed change relocates surveillance frequencies to a licensee-controlled program, the SFCP. This change is applicable to licensees using probabilistic risk guidelines contained in NRC-approved NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," (ADAMS Accession No. ML071360456).

Basis for proposed no significant hazards consideration: As required by 10 CFR 50.91(a), the PSEG analysis of the issue of no significant hazards consideration is presented below:

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

   Response: No

   The proposed change relocates the specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the technical specifications for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   No new or different accidents result from utilizing the proposed change. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No

   The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to TS), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, PSEG will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Rev. 1 in accordance with the TS SFCP. NEI 04-10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above, PSEG concludes that, the requested change does not involve a significant hazards consideration as set forth in 10 CFR 50.92(c), "Issuance of Amendment."