NEXTera ENERGY SEABROOK

May 28, 2013

10 CFR 50.90

SBK-L-13071
Docket No. 50-443

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

Seabrook Station

License Amendment Request 13-03

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

In accordance with the provisions of Title 10 of the Code of Federal Regulations (10 CFR Part 50.90), "Application for Amendment of License, Construction Permit, or Early Site Permit," NextEra Energy Seabrook, LLC (NextEra) is submitting a request for an amendment to the Technical Specifications (TS) for Seabrook Station. The proposed amendment would modify Seabrook's TS by relocating specific surveillance frequencies to a licensee-controlled program with implementation of Nuclear Energy Institute (NEI) 04–10, "Risk-Informed Technical Specification Initiative 5B, Risk-Informed Method for Control of Surveillance Frequencies."

The changes are consistent with NRC-approved 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 changes, the requested confirmation of applicability, and plant-specific verifications. Attachment 2 provides documentation of PRA technical adequacy. Attachment 3 provides the existing TS pages marked up to show the proposed changes, and Attachment 4 provides the proposed TS Bases changes. The changes to the TS Bases are provided for information only and will be incorporated in accordance with the TS Bases Control Program upon implementation of the approved amendment. Attachment 5 contains the proposed No Significant Hazards Consideration. Attachment 6 provides a cross-reference between the surveillance requirements (SR) contained in TSTF-425, "Relocate Surveillance Frequencies to Licensee Control—RITSTF Initiative 5b," and the SR in the Seabrook Station TS.

A001
NRR

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 New Hampshire State Liaison.

This letter contains no regulatory commitments.

NextEra requests NRC review and approval of LAR 13-03 with issuance of a license amendment by June 1, 2014 and implementation of the amendment within 90 days.

Should you have any questions regarding this letter, please contact Mr. Michael O'Keefe, Licensing Manager, at (603) 773-7745.

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

Executed on May 28, 2013.

Sincerely,

Kevin T. Walsh
Site Vice President
NextEra Energy Seabrook, LLC

Attachments:

1. Description and Assessment of Proposed Changes
2. Documentation of PRA Technical Adequacy
3. Mark-up of Proposed Technical Specification Changes
4. Mark-up of Proposed Technical Specification Bases Changes
5. No Significant Hazards Consideration
6. Cross-reference of TSTF 425 and Seabrook Surveillance Requirements

cc: NRC Region I Administrator
J. G. Lamb, NRC Project Manager
NRC Senior Resident Inspector

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Attachment 1

**Description and Assessment**

License Amendment Request 13-03

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

## 1.0 DESCRIPTION

The proposed amendment would modify the Seabrook Station Technical Specifications (TS) by relocating specific surveillance frequencies to a licensee-controlled program with the adoption of Technical Specification Task Force (TSTF)–425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control—Risk Informed Technical Specification 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 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

NextEra Energy Seabrook, LLC (NextEra) 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 (ADAMS Accession No. ML071360456) [Reference 3].

Attachment 2 includes NextEra's documentation with regard to PRA technical adequacy consistent with the requirements of Regulatory Guide 1.200, Revision 1 (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.

NextEra has concluded that the justifications presented in the TSTF proposal and the safety evaluation prepared by the NRC staff are applicable to Seabrook Station and justify this amendment to incorporate the changes to the Seabrook TS.

### 2.2 Optional Changes and Variations

The proposed amendment is consistent with the STS changes described in TSTF-425, Revision 3, but NextEra 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 that may affect 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 Seabrook TS were based on NUREG-0452, Standard Technical Specifications - Westinghouse Pressurized Water Reactors. As a result, the Seabrook TS surveillance numbers and associated Bases numbers differ from the surveillance and Bases numbers in NUREG-1431 and TSTF-425, Revision 3. In addition, the Administrative Controls section of the TS is Section 6.0 for Seabrook 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 Seabrook TS. Therefore, the NUREG-1431 mark-ups included in TSTF-425 for these surveillances are not applicable to Seabrook. 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).

3. The Seabrook 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. NextEra has determined that the relocation of the frequencies for these Seabrook-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 [Reference 5], relative to changes in surveillance frequencies. Therefore, the proposed relocation of the Seabrook-specific surveillance frequencies is consistent with TSTF-425 and with the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

4. The definition of STAGGERED TEST BASIS is being retained in Seabrook TS Definition Section 1.0 since this terminology is used in Administrative TS Section 6.7.6.l, "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).

5. The insert provided in TSTF-425 for the TS Bases (Insert 2) states: "The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program." In a letter dated April 14, 2010 [Reference 6], the NRC staff agreed that the insert applies to surveillance frequencies that are relocated and subsequently evaluated and changed in accordance with the SFCP, but it does not apply to frequencies relocated to the SFCP but not changed. Therefore, the insert for the bases is revised to: "The Surveillance Frequency is controlled under the Surveillance Frequency Control Program." 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).

6. The Seabrook TS, which were based on NUREG-0452, do not contain Bases as comprehensive as those in NUREG-1431, which discuss most all SR. Therefore, many of the Bases mark-ups in TSTF-425 are not applicable to the Seabrook TS. The proposed Bases changes in Attachment 4 revise only those Bases that currently discuss surveillance frequencies. 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 existing Bases information describing the basis for the surveillance frequencies will be relocated to the Seabrook Surveillance Frequency Control Program.

7. The SR for reactor trip and engineered safety features actuation system instrumentation in Seabrook TS 3.3.1 and 3.3.2 are presented in a tabular format, which is different from the format of the SR for the same instrumentation in NUREG-1431. To accommodate this difference, the proposed change includes use of "SFCP" as a frequency notation in the tables that specify instrumentation SR. This is an administrative deviation from TSTF-425 due to differences in format between the Seabrook TS and NUREG-1431, which has no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

8. Surveillance requirement (SR) 4.3.3.1 requires that "Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3." The requirement to demonstrate operability of instrument channels "for the

MODES and at the frequencies shown in Table 4.3-3" is being deleted from SR 4.3.3.1. This requirement is not a restriction on the MODES in which the SRs are required. The MODES shown in Table 4.3-3, Radiation Monitoring Instrumentation for Plant Operations Surveillance Requirements, are identical to the applicable MODES for which the instrument channels are required to be OPERABLE as shown in Table 3.3-6, Radiation Monitoring Instrumentation for Plant Operations. SR 4.0.1 requires SR to be met during the MODES or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the SR. Therefore, the reference to MODES in SR 4.3.3.1 is redundant to the requirements of SR 4.0.1.

Further, with the proposed change to SR 4.3.3.1, listing the SR frequencies in Table 4.3-3 is redundant to SR 4.3.3.1. Therefore, this proposed change deletes Table 4.4-3 since all of the information provided in the table is included elsewhere within TS 3.3.3.1. The SR and frequencies in Table 4.3-3 are incorporated in the proposed change to SR 4.3.3.1. The information in Table 4.3-3 column "Modes for Which Surveillance is Required," including the associated footnotes, is also contained in Table 3.3-6 in the column "Applicable Modes" and in the TABLE NOTATIONS. Therefore, eliminating Table 4.3-3 does not remove any requirements but only eliminates duplicate information.

The same change (deletion of Table 4.3-3) was approved for Salem Units 1 and 2 in Amendments 299 and 282 [Reference 7], which adopted TSTF-425. This change aligns the Seabrook 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.

9. TSTF-425 excludes relocating frequencies that reference other approved programs for the specific interval (such as the Inservice Testing Program or the Primary Containment Leakage Rate Testing Program). The approved programs for Seabrook are described in Section 6.0, "Administrative Controls," of the Seabrook TS. The title of TS Table 4.4-3 (Reactor Coolant Specific Activity Sample and Analysis Program) may be misconstrued as a program; however, Section 6.0 of the TS does not contain a program for sampling and analysis of reactor coolant specific activity. To avoid a misunderstanding of these SR, NextEra proposes to delete the word "Program" from the title of TS Table 4.4-3. Consistent with TSTF-425, the eligible frequencies in Table 4.4-3 are proposed for relocation to the Surveillance Frequency Control Program. This change aligns the Seabrook 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

NextEra has reviewed the proposed no significant hazards consideration determination (NSHC) published in the Federal Register July 6, 2009 (74 FR 31996). NextEra has concluded that the proposed NSHC presented in the Federal Register notice is applicable to Seabrook 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). NextEra has concluded that the relationship of the proposed changes to the applicable regulatory requirements presented in the Federal Register notice is applicable to Seabrook.

### 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

NextEra 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). NextEra concluded that the staff's findings presented therein are applicable to Seabrook, 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 Number: ML090850642).

2. 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; July 6, 2009 (74 FR 31996)

3. NEI 04-10, Revision 1, "Risk-Informed Method for Control of Surveillance Frequencies," April 2007 (ADAMS Accession Number: ML071360456)

4. Regulatory Guide 1.200, Rev. 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1, January 2007 (ADAMS Accession Number: ML070240001)

5. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications," August 1998 (ADAMS Accession Number ML003740176)

6. NRC Letter to Technical Specifications Task Force, "Notification of Issue with NRC-approved Technical Specifications Task Force (TSTF) Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control -RITSTF Initiative 5b," April 14, 2010

7. NRC Letter "Salem Nuclear Generating Station, Unit Nos 1 and 2, Issuance of Amendments RE: Relocation of Specific Surveillance Frequencies to a Licensee-Controlled Program Based on Technical Specifications Task Force (TSTF) Change TSTF-425 (TAC Nos. ME3574 and ME3575)," March 21, 2011

<u>Attachment 2</u>

Documentation of PRA Technical Adequacy

## Documentation of PRA Technical Adequacy

### *Table of Contents*

## 1.0 INTRODUCTION

The implementation of the Surveillance Frequency Control Program (SFCP, also referred to as Technical Specifications Initiative 5b) at Seabrook Station will follow the guidance provided in NEI 04-10, Revision 1 in evaluating proposed surveillance test interval (STI) changes. The following steps of the risk-informed STI revision process are common to all proposed STI changes within the proposed licensee controlled program.

- Each proposed 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 can proceed. If a commitment exists and the commitment change process does not permit the change without NRC approval, then the STI revision cannot be implemented. Only after receiving NRC approval to change the commitment could a STI revision proceed.

- A qualitative analysis is performed for each STI revision that involves several considerations as explained in NEI 04-10, Revision 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 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 documented, implemented, and available 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. Performance monitoring helps to confirm that no failure mechanisms related to the revised test interval are subsequently identified as sufficiently significant to alter the basis provided in the justification for the surveillance interval change.

- 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 Probabilistic Risk Assessment (PRA) is used, when possible, to quantify the effect of a proposed individual STI revision compared to acceptance criteria in NEI 04-10, Revision 1. Also, the cumulative impact of risk-informed STI revisions on PRA evaluations (i.e., internal events, external events, and shutdown) is compared to the risk acceptance criteria as delineated in NEI 04- 10, Revision 1. 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.

The NEI 04-10, Revision 1 methodology endorses the guidance provided in Regulatory Guide (RG) 1.200, Revision 1, "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.

    - Identify structures, systems, and components (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 events, external events, applicable modes), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the PRA 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 RG (currently, RG 1.200, Revision 1, which includes only the 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 attachment is to address the requirements identified in item 4 above.

This evaluation summarizes the assessment of the Seabrook PRA capability as measured against the current ASME/ANS PRA Standard (ASME/ANS RA-Sa-2009) [Reference 1], endorsed by NRC Regulatory Guide (RG) 1.200, Revision 2 [Reference 2]. While the NEI guidance document refers to RG 1.200, Revision 1, which includes only the internal events PRA standard (ASME RA-Sb-2005), this evaluation addresses the broader scope of RG 1.200, Revision 2. This assessment addresses the technical adequacy of the Seabrook PRA for use in risk-informed applications.

The assessment is based on a series of formal peer reviews and other technical reviews, documented in the EPRI ePSA database tool [Reference 3] and summarized in Tables 1 and 2. This assessment uses the latest Seabrook PRA, SSPSS-2011 Update [Reference 4], which incorporated a significant update to internal flood analysis along with a number of miscellaneous changes.

## 2.0 BACKGROUND

### *2.1 RG1.200 & PRA Standard*

The ASME / ANS PRA Standard (ASME/ANS RA-Sa-2009) has eight "parts" with technical elements, high level requirements (HLRs), and detailed supporting requirements (SRs). These parts represent the major classes of hazards included in a PRA:

- Part 2, internal events (addressed in Section 3.1),
- Part 3, internal flood (addressed in Section 3.2),
- Part 4, internal fire (addressed in Section 3.3),
- Part 5, seismic events (addressed in Section 3.4),
- Parts 6 to 9, other external hazard events (addressed in Section 3.5).

Note, Part 1 of the PRA Standard is introductory information and does not contain any requirements except configuration control (addressed in Section 3.6).

NRC Reg Guide 1.200 Rev 2 endorses this Standard with minor "clarifications." The EPRI ePSA database includes each supporting requirement from ASME/ANS RA-Sa-2009 along with the clarifications from NRC Reg Guide 1.200 Rev 2.

The Standard supporting requirements allow the assessment of the portions of the PRA as Capability Category CC-I, CC-II, or CC-III, with increasing scope and level of detail, plant-specificity, and realism. Thus, the overall assessment of PRA capability is the collection of the assessments of the hundreds of supporting requirements.

## 2.2 Seabrook PRA History

Table 3 summarizes the history of the Seabrook PRA, from the original PRA in 1983 to the current 2011 update. The Seabrook PRA has been maintained during that time as a living PRA.

The Seabrook PRA was originally developed in 1983 as a full-scope Level 3 risk assessment of at-power operation of Seabrook (SSPSA-1983). This PRA was subject to a number of reviews, internal and external, during its preparation as well as extensive review by NRC through national labs following its publication.

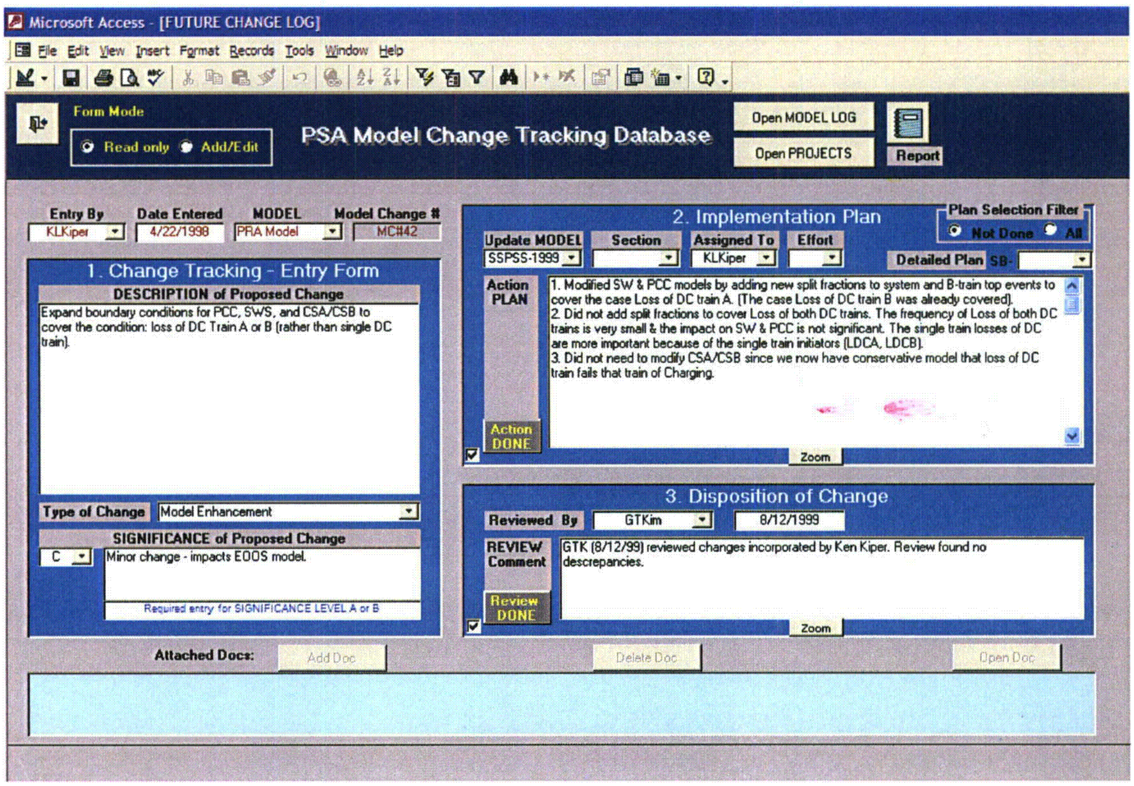
The Seabrook PRA has been revised and upgraded a number of times since the original SSPSA-1983, including the 1990 update which formed the basis for the IPE submittal and the 1993 update which formed the basis for the IPEEE submittal.

The 1999 PRA was the first revision that was subject to the industry PSA certification process, the precursor to the current PRA Standard. Since that time, the Seabrook PRA has been subject to a number of peer reviews, self assessments, and other outside expert reviews.

The current model of record is the SSPSS-2011 (Seabrook Station Probabilistic Safety Study, 2011 Update, Reference 4). It is a full-scope Level 2 risk assessment of all modes of operation at Seabrook Station.

## 2.3 Model Change Database

The living PRA is maintained through use of a Model Change Database (MCDB). A sample screen shot of the input form is shown below.



This MCDB is used to track all open items against the current PRA. This includes all findings and observations from peer reviews, external reviews, and self assessments, as well as issues identified during use and update of the PRA model.

Tables 1 and 2 provide a summary of MCDB records, as they are associated with specific reviews. This table summarizes the number of the findings and observations, graded as significance level A, B or C/D, and whether the finding was closed or remains open. The significance levels are used to assess each finding for its potential impact on the PRA model. The significance levels are defined as follows:

- **"A" Finding:** MAJOR model weakness. *Extremely important and necessary to address to ensure the technical adequacy of the PSA, the quality of the PSA or the quality of the PSA update process.*
- **"B" Finding:** IMPORTANT plant or model change or model weakness. *Important and necessary to address but may be deferred until the next PSA* update.
- **"C" Observation:** MINOR plant or model change or model error. *Considered desirable to maintain maximum flexibility in PSA applications and consistency in the industry, but not likely to significantly affect results or insights.*
- **"D" Observation:** DOCUMENTATION Change Only. *Editorial or minor technical item left to the discretion of the host utility.*

As these tables show, of a total of 155 + 739 = 894 records, there are currently 14 + 107 = 121 remaining open items, all of which are C/D level observations. Most of these open items are classified as "enhancements" or "documentation only." Since this is a living PRA, this number will change over time. Additional open items will be identified in the future. Open items will be addressed in future PRA updates, based on the significance of the open item and the scope of the update.

Significant findings ("A" and "B" level) from the peer reviews are listed in Attachment A, along with their resolutions.

### *2.4 Seabrook PRA Capability Target*

The target capability level for the Seabrook PRA is Capability Category II (CC-II). That is, the goal is to meet all supporting requirements at least at the CC-II level. This is the maximum capability level needed by any foreseeable application. Any supporting requirement identified as "Not Met" or "Meets CC-I only" would include an associated finding.

Note that in many supporting requirements, the requirement spans all three capability categories. Thus, if the SR is met, it meets CC III. While CC II is the target, CC III is met in many SRs.

### 2.5 Assessment Process

The assessment of PRA capability judges the Seabrook PRA against each supporting requirement in the PRA Standard as "Meets" CC-I, CC-II, or CC-III. If the PRA does not meet the requirements of any category for a specific SR, it is assessed as "Not Met." In some cases where the SR is substantially met, it may be assessed as "Met with Exception" with an open item to track the exception.

This assessment is captured in two databases:

- The SR-by-SR assessments from industry peer reviews and internal self assessments are documented in the EPRI ePSA Version 3.2 database tool.
- Any finding or observation from a peer review or self assessment is entered into the Seabrook Model Change database for tracking, as described in Section 2.3 above. The Model Change database record number is used in the ePSA database to link open items for requirements "Not Met" or "Met with Exception" to the specific activity needed to address the issue.

The most recent self assessment (SA-2011) addressed the open items from previous reviews that were closed in the 2011 PRA Update. A subsequent focused peer review was performed in 2012 for element LE that identified additional open items. The remaining open items will be addressed in future PRA updates.

## 3.0 EVALUATION

The following sections describe the capability of the Seabrook PRA for the major Standard parts.

### 3.1 Part 2 Internal Events

The internal events portion of the Seabrook PRA has been updated a number of times since the original SSPSA-1983, including the most recent SSPSS-2011 Update.

Three peer reviews have been conducted against internal event supporting requirements:

- In 1999, a review of all technical elements was performed using the industry PSA Certification process, the precursor to the PRA Standard. The 1999 peer review resulted in 74 findings and observations (2 "A" level, 28 "B" level and 44 "C/D" level). All of the findings and observations have been addressed in the 2011 Update (or prior updates).
- In 2005, a focused peer review was performed for the elements AS, SC and HR as well as configuration control. This assessment replaced the 1999 peer review for those elements that were in scope. This review was done using the then current PRA Standard (ASME RA-Sa-2003). The 2005 focused peer review resulted in 42 findings and observations (4 "B" level and 38 "C/D" level). All but one of the findings and observations (a "C/D" level F&O) have been addressed in the 2011 Update (or prior updates).
- In 2012, a focused peer review was performed for the element LE. This assessment replaced the 1999 peer review for that element. This review was

done using the current PRA Standard (ASME/ANS RA-Sa-2009). The 2012 peer review resulted in 13 findings and observations (6 findings, 7 suggestions), along with one best practice. These findings and observations were reviewed and classified as "C" level (9) and "D" level (4). These issues will be addressed in a future update.

Three external reviews were conducted in 2002, 2006, and 2011 of specific internal events model issues. While these reviews were not specifically against PRA Standard requirements, they represented outside technical reviews which identified observations for improvement. A total of 50 "C/D" level observations were identified; all but one of these observations have been addressed and closed.

Four self assessments against the internal event SRs in the PRA standard were performed in 2005 (ASME RA-Sa-2003), 2007 (ASME RA-Sb-2005), 2010 (ASME/ANS RA-Sa-2009) and 2011 (ASME/ANS RA-Sa-2009). The first three self assessments considered all internal events technical elements. The SA-2011 addressed only the open findings against specific SRs. These assessments identified 33 "C/D" level findings. Of these, 4 findings remain open.

The 2011 Self Assessment represents the most current status of Seabrook PRA capability, except for element LE. It updated the 2010 Self Assessment, based on open items closed out in the 2011 PRA Update. The 2010 Self Assessment had assessed the 2009 PRA against each of the 254 internal events supporting requirements in ASME/ANS RA-Sa-2009. That assessment reviewed the results of previous peer reviews and their observations along with the subsequent revisions to the PRA that addressed the observations

The 2011 assessment is documented in the EPRI ePSA database, ePSA_2011.ssa. The following table summarizes the results of the 2011 self assessment and the 2012 focused peer review:

***Number of Supporting Requirements that Meet CC I, II, III for Internal Events PRA***

| | | Meet Capability Category | | | |
|---|---|---|---|---|---|
| **Technical Element** | **Not Met** | **CC-I** | **CC-II** | **CC-III** | ***Total*** |
| IE Initiating Events | 0 | 0 | 10 | 23 | *33* |
| AS Accident Sequences | 0 | 0 | 1 | 20 | *21* |
| SC Success Criteria | 0 | 0 | 0 | 14 | *14* |
| SY Systems Analysis | 0 | 0 | 4 | 38 | *42* |
| HR Human Reliability Analysis | 0 | 0 | 3 | 32 | *35* |
| DA Data Analysis | 0 | 0 | 4 | 29 | *33* |
| QU Quantification | 0 | 0 | 1 | 34 | *35* |
| LE LERF Analysis | 2 | 4 | 9 | 26 | *41* |
| *Total* | *2* | *4* | *32* | *216* | *254* |

Thus, of 254 supporting requirements for internal events (Part 2), all but six are assessed Capability Category II or III (most III). The six supporting requirements that are not met at CC II or III are documented in Attachment B, along with an evaluation of their significance. The conclusion is that none of these open items represents a significant deficiency in the LERF analysis and that their future resolution will enhance the documentation but will not affect the LERF total or impact the significant LERF contributors.

Conclusion

With the exception of the LE supporting requirements identified above, the 2011 PRA meets all Part 2 (internal event) CC II requirements of the PRA Standard.

### *3.2 Part 3 Internal Flood*

The internal flood portion of the Seabrook PRA (IF PRA) has been updated several times, including the most recent 2011 revision of the 1991 IPEEE submittal (which was an update of the original SSPSA-1983). This current update uses the most recent technical guidance for internal flood analysis.

A peer review was conducted in 2009 of the IF PRA, using two industry experts. The peer review resulted in a total of 32 findings and observations, of these were 16 findings, 14 suggestions, and 2 best practices. The findings and suggestions were later grouped into a set of 26 F&Os (3 "B" level and 23 "C/D" level) and were entered into the Model Change database. The peer review identified five supporting requirements that were not met. These five SRs included: IFSO-A4 (F&O 5-13), IFQU-A7 (F&O 4-6), IFQU-B1 (F&O 4-7), IFQU-B2 (F&O 4-8) and IFQU-B3 (F&O 4-9).

A self assessment of the internal flood PRA was conducted in 2010 following the revision to the IF PRA that addressed the 2009 Peer Review F&Os. This self assessment addressed each of the 62 internal flood supporting requirements, based on a review of the 2009 Peer Review and an assessment of changes to the IF PRA. This 2010 SA identified no new observations or findings and determined that all IF SRs were met, with two exceptions (IFQU-A7 and IFQU-B1). These final open items were closed in the 2011 PRA Update.

The 2011 self assessment updated the 2010 self assessment, based on changes made in the 2011 PRA Update. The following table summarizes the results of the 2011 self assessment:

***Number of Supporting Requirements that Meet CC I, II, III for Internal Flood PRA***

| | | Meet Capability Category | | | |
|---|---|---|---|---|---|
| **IF Technical Element** | **Not Met** | **CC-I** | **CC-II** | **CC-III** | ***Total*** |
| IFPP IF Plant Partitioning | 0 | 0 | 0 | 8 | *8* |
| IFSO IF Source Identification | 0 | 0 | 0 | 9 | *9* |
| IFSN Scenario Development | 0 | 0 | 3 | 17 | *20* |
| IFEV IF-Induced IE Analysis | 0 | 0 | 2 | 9 | *11* |
| IFQU IF Quantification | 0 | 0 | 1 | 13 | *14* |
| *Total* | *0* | *0* | *6* | *56* | *62* |

Thus, all of the 62 supporting requirements for IF assessed Capability Category II or III (most III).

Conclusion

The 2011 PRA fully meets all Part 3 (internal flood) CC II requirements of the PRA Standard.

### *3.3 Part 4 Internal Fire*

The internal fire portion of the Seabrook PRA (FPRA) has been updated several times, including the most recent 2004 revision of the 1991 IPEEE submittal (which was an update of the original SSPSA-1983). This recent update used then-current methods, but was performed before the most recent technical guidance in NUREG/CR-6850.

The Seabrook Fire PRA has not been subject to self assessment or peer review, but was independently reviewed by external experts in 2004 (see Table 2). Because of the significant methodological revision since the 2004 update, a self assessment against the PRA Standard would not be effective.

Conclusion

The FPRA was performed and revised using qualified analysts with then-current methods and it is believed to be a reasonable representation of the fire risk from operation of Seabrook Station.

### *3.4 Part 5 Seismic Events*

The seismic events portion of the Seabrook PRA (SPRA) has been updated several times, including the most recent 2005 revision of the 1991 IPEEE submittal (which was an update of the original SSPSA-1983). This recent update used current methods to address issues related to equipment and operator fragility and the revised hazard spectrum but it did not include an update to the seismic hazard curve.

The Seabrook Seismic PRA has not been subject to self assessment or peer review, but was independently reviewed by external experts in 2005 (see Table 2).

Conclusion

The SPRA was performed and revised using qualified analysts with then-current methods and it is believed to be a reasonable representation of the seismic hazard risk from operation of Seabrook Station.

### *3.5 Parts 6 to 9 Other External Hazards*

The analysis of other hazards was performed for the original SSPSA-1983 and was updated for the 1991 IPEEE report. This portion of the Seabrook PRA has not been subject to self assessment or peer review.

Conclusion

The Other Hazards PRA was performed and revised using qualified analysts with then-current methods and it is believed to be a reasonable representation of the other hazards risk from operation of Seabrook Station.

*3.6 PRA Model Maintenance & Control*

PRA model maintenance and control requirements are described in the PRA Standard Section 1-5. These requirements were assessed in the 2005 peer review, which resulted in observations MC#542 through 553. These observations have each been addressed in the current set of department instructions that address model maintenance and control:

PRA-101, PRA Configuration Control

PRA-102, PRA Model Maintenance

PRA-103, Control of Risk Informed Applications

PRA-104, Control of PRA Computer Codes

PRA-105, SB PRA Peer Reviews

PRA-106, PRA Modeling Guidelines

## 4.0 CONCLUSION

The Seabrook PRA model of record (SSPSS-2011) fully meets all the requirements of Part 2 "Internal Events"(with minor exceptions for element LE) and Part 3 "Internal Flood" of the ASME/ANS PRA Standard. All significant findings (level "A" or "B") from peer reviews or other technical reviews have been addressed and closed.

The fire PRA, seismic PRA, and other hazards PRA portions of the Seabrook PRA have been maintained and have been subject to independent review by external experts. However, these portions of the PRA have not been assessed against the requirements of the PRA Standard.

## 5.0 REFERENCES


1. ASME/ ANS RA-Sa-2009, "Addenda to ASME/ ANS RA-S-2008 Standard for Level 1/ Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", American Society of Mechanical Engineers and American Nuclear Society, 2009.
2. U.S. Nuclear Regulatory Commission, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Regulatory Guide 1.200, Revision 2, 2008.
3. EPRI ePSA Database, Version 3.2.
4. Seabrook Station Probabilistic Safety Study, 2011 Update (SSPSS-2011), June 2011.
5. Seabrook Engineering Evaluation EE-11-026, "Seabrook PRA Capability Assessment," October 2011.

## *TABLE 1 Summary of Peer Reviews (1999 to 2012)*

| Review ID | Reviewers | Standard | PRA Model | Scope of Review | | Findings & Observations [(a)] TOTAL (OPEN) | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | | PRA Scope | STD Elements | A | B | C/D | *Total* |
| *1999 Peer Review* | B. Sloane, M. Averett, F. Cietek, K. Connelly, P. Guymer, R. Lutz, S. Rodgers, Y. Khalil | Industry PSA Certification Guideline Document (Rev A3) | SSPSS-1999 | Internal events + internal flood, full power | ALL elements | 2 (0) | 28 (0) | 44 (0) | *74 (0)* |
| *2005 Focused Peer Review (AS, SC, HR)* | J. Moody, J. Bretti, D.Gaynor | ASME RA-Sa-2003 | SSPSS-2004 | Internal events, full power | AS, SC, HR (E to I), Configuration Control | 0 (0) | 4 (0) | 38 (1) | *42 (1)* |
| *2009 Focused Peer Review (IF)* | R. Kirchner, C. Guey | ASME/ANS RA-Sa-2009 | SSPSS-2009 + 2010 IF Update | Internal flood, full power | ALL IF elements | 0 (0) | 3 (0) | 23 (0) | *26 (0)* |
| *2012 Focused Peer Review (LE)* | E. Thornsbury, J. Moody, M. Averett | ASME/ANS RA-Sa-2009 | SSPSS-2011 | Internal events, full power | LE | 0 (0) | 0 (0) | 13 (13) | *13 (13)* |
| ***TOTAL*** | | | | | | **2 (0)** | **35 (0)** | **118 (14)** | ***155 (14)*** |

NOTES:

(a) The Findings & Observations (F&Os) are ranked as follows (see instruction PRA-101 PRA Configuration Control):

- "A" Finding: MAJOR model weakness. Extremely important and necessary to address to ensure the technical adequacy of the PSA, the quality of the PSA or the quality of the PSA update process.
- "B" Finding: IMPORTANT plant or model change or model weakness. Important and necessary to address but may be deferred until the next PSA update.
- "C" Observation: MINOR plant or model change or model error. Considered desirable to maintain maximum flexibility in PSA applications and consistency in the industry, but not likely to significantly affect results or insights.
- "D" Observation: DOCUMENTATION Change Only. Editorial or minor technical item left to the discretion of the host utility.

## *TABLE 2 Summary of Other Technical Reviews (1999 to 2012)*

| Review ID | Reviewers | Standard | PRA Model | Scope of Review | | Findings & Observations TOTAL (OPEN) | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | | PRA Scope | STD Elements | A | B | C/D | *Total* |
| *2002 External Review (RM)* | D. Wakefield | None (good practice for RISKMAN modeling) | SSPSS-2002 | Internal events & internal flood, full power | (model structure) | 0 (0) | 0 (0) | 35 (0) | *35 (0)* |
| *2002 External Review (LPSD)* | J.Moody, R. Kirchner | None (reviewer expert judgment for LPSD model) | SSPSS-2002 | Internal events, low power & shutdown | (model structure, IEs, HRA, results) | 0 (0) | 1 (0) | 0 (0) | *1 (0)* |
| *2004 External Review (FPRA)* | A. Afzali, C.Guey | None (reviewer expert judgment for Fire PRA model) | SSPSS-2004 | Internal fire PRA, full power | n/a | 0 (0) | 0 (0) | 0 (0) | *0 (0)* |
| *2005 Self Assessment* | K.Kiper | ASME RA-Sa-2003 | SSPSS-2004 | Internal events & internal flood, full power | ALL elements | 0 (0) | 0 (0) | 27 (3) | *27 (3)* |
| *2005 External Review (SPRA)* | J.Moody, B. Campbell | ANS 58.21-2003 | SSPSS-2004 | Seismic events, full power | ALL seismic elements | 0 (0) | 1 (0) | 2 (1) | *3 (1)* |
| *2006 External Review (RM)* | D. Wakefield | None (good practice for RISKMAN modeling) | SSPSS-2005 | Internal events, full power | (ac power model) | 0 (0) | 0 (0) | 14 (0) | *14 (0)* |
| *2007 Self Assessment* | K. Kiper | ASME RA-Sb-2005 | SSPSS-2006 | Internal events, full power | ALL elements | 0 (0) | 0 (0) | 5 (1) | *5 (1)* |
| *2010 Self Assessment* | K. Kiper, G. Kim, R. Turcotte | ASME/ANS RA-Sa-2009 | SSPSS-2009 | internal events, full power | ALL elements | 0 (0) | 0 (0) | 1 (0) | *1 (0)* |
| *2010 Self Assessment (IF)* | K. Kiper, R. Turcotte | ASME/ANS RA-Sa-2009 | SSPSS-2010 IF Update | internal flood, full power | ALL IF elements | 0 (0) | 0 (0) | 0 (0) | *0 (0)* |
| *2011 External Review (L2)* | J.Gabor | n/a | SSPSS-2011 | Internal events, full power | (Level 2 release category definitions and associated source terms) | 0 (0) | 0 (0) | 1 (1) | *1 (1)* |
| *2011 Self Assessment* | K. Kiper, R. Turcotte | ASME/ANS RA-Sa-2009 | SSPSS-2011 | Internal events & internal flooding, full power | ALL elements (addressed SRs with open items) | 0 (0) | 0 (0) | 0 (0) | *0 (0)* |
| *Other Internal Reviews* [(a)] | K. Kiper, G. Kim, R.Turcotte, | n/a | All | all | n/a | 0 (0) | 58 (0) | 594 (101) | *652 (101)* |
| ***TOTAL*** | | | | | | **0 (0)** | **60 (0)** | **679 (107)** | ***739 (107)*** |

NOTES:
(a) Other Internal Reviews – these findings were internally generated during periodic model updates or use of the PRA models. These findings included model enhancements, model errors, plant changes, and documentation changes. All "B" review comments have been addressed, along with most of the other comments.

## *TABLE 3 History of the Seabrook PRA*

| UPDATE ID | DATE | RM MODEL | CDF | UPDATE Description |
|---|---|---|---|---|
| SSPSS-2011 | Jun-11 | SB2011X | 1.23E-05 | FOCUSED update to address open items to conform to RG1.200 Rev 2, including revised internal flood analysis and latent human failure events analysis. |
| SSPSS-2009 | Jun-09 | SB2009X | 1.17E-05 | PERIODIC update to incorporate plant-specific data, latest generic data, operator input into HRA, and a number of model enhancements. |
| SSPSS-2006 | Jan-07 | SB2006X | 1.45E-05 | FOCUSED update to incorporate changes in the shutdown PRA model based on insights from outage risk management during OR11. |
| SSPSS-2005 | Aug-06 | SB2005X | 1.36E-05 | MAJOR update to incorporate updated success criteria and sequence timing, seismic PRA revision, Level 2 model revision, and a number of model enhancements. |
| SSPSS-2004 | Dec-04 | SB2004X | 2.99E-05 | MAJOR update to incorporate SEPS, enhancements to HRA, system modeling, sequence documentation, and fire PRA. |
| SSPSS-2002 | Jan-03 | SB2002X | 4.51E-05 | MAJOR update, addressed peer review comments, data |
| SSPSS-2001 | Oct-01 | SB2001 | 4.79E-05 | MINOR update to incorporate changes to support export to Safety Monitor, updated initiating event modeling. |
| SSPSS-2000 | Jan-01 | SB2000 | 4.63E-05 | Minor update to incorporate new RISKMAN Windows version. Restructured modeling of train-related top events. |
| SSPSS-1999 | May-00 | SB99PR | 4.57E-05 | MAJOR update, top event modeling and incorporated plant changes, data. |
| SSPSS-1996a | Mar-98 | SB98 | 4.32E-05 | MINOR changes to system modeling & HRA values. |
| SSPSS-1996 | Dec-97 | SB9701 | 4.24E-05 | MAJOR update, IE models and data. |
| SSPSS-1993a | Mar-95 | SB9501 | 7.20E-05 | MINOR update - software and documentation update |
| SSPSS-1993 | Sep-93 | --- | 8.00E-05 | MAJOR update to support IPEEE, data. |
| SSPSS-1990a | Sep-92 | --- | 9.40E-05 | MAJOR update - external events analysis update only. |
| SSPSS-1990 | Mar-91 | --- | 1.10E-04 | MAJOR update - to support IPE submittal. |
| SSPSS-1989 | May-90 | --- | 1.10E-04 | MAJOR update, update to IE and CCF modeling. |
| SSPSS-1986 | Jul-87 | --- | 2.70E-04 | MAJOR update. (Renamed the updated analysis "SSPSS" to distinguish from the original SSPSA.) |
| SSPSA (1983) | Dec-83 | --- | 2.30E-04 | Original at-power PRA to assess the baseline risk from Seabrook Station operation. |

## *Attachment A Significant Findings from Peer Reviews (Closed)*

This table summarizes findings and observations with significance ranking "A" or "B" from three peer reviews (1999, 2005, 2009) listed in Table 1. The 2012 focused peer review for LE had no significant F&Os. Note that the focused peer reviews for 2005 (AS, SC, HR) and 2012 (LE) replace the assessments made for those elements in the 1999 peer review, and, as a result, the AS, SC, HR & LE findings from the 1999 peer review are not listed in this table.

All of the significant findings have been addressed, as documented in the Resolution column, and are closed. The first column (MC ID) is the number used in the Model Change database to track each finding and its resolution.

| MC ID | PR | F&O ID | Peer Review Finding | Resolution |
|---|---|---|---|---|
| MC#125 | 1999 Peer Review (OG Cert) | F&O IE-2 (IE-C10) | The frequencies of initiators L2CCA and L2CCB are under estimated due to the common cause model. The common cause term should include T=1 year (rather than 24 hours). | Changes were made to the CCF models in PCC and SWS initiators to use 1 year as the mission time.<br>- Included in 1999 PRA Update. |
| MC#129 | 1999 Peer Review (OG Cert) | F&O IE-6 (IE-C14) | The existing analyses for ISLOCA should be reviewed for consistency with a methodology for identification and quantification of ISLOCA pathways such as that provided in NUREG/CR-5744, and updated if appropriate. | Reviewed NUREG/CR-5744 for ISLOCA methodology and revised the ISLOCA assessment.<br>- Included in 2005 PRA Update. |
| MC#133 | 1999 Peer Review (OG Cert) | F&O AS-6 (DA-C15) | The emergency diesel generator recovery failure probability seems optimistic for the medium RCP seal LOCA event. The data for recovery of an EDG is based on data taken from LERs based on EDG failures. This data is used to develop a recovery curve. However, this recovery is applied in conditions very different than the conditions in the LER - common cause failure of both EDGs resulting in SBO conditions. The EDG recovery is based on generic data composed of EDG single failures during normal operation. This data needs to be reviewed to ensure applicability to CCF events, particularly events during more adverse SBO conditions (i.e., where stress, crew availability, and so forth, are more limiting). In addition, plant-specific evidence should be used to support this recovery probability. | Evaluated Seabrook Station EDG failure data. Of the four failures, two could easily be recovered within 4 hours. The other two failures were considered long-term failures. Based on SB data, a non-recovery probability of 0.5 was used for DG recovery.<br>- Included in 2002 PRA Update. |
| MC#163 | 1999 Peer Review (OG Cert) | F&O DA-4 (DA-D5) | The values for BETA2, GAMMA2, and DELTA2 are not derived as recommended in NUREG/CR-5485 as stated in the text. That document (p.76) recommends that "the values of $\alpha_2$, $\alpha_3$, and $\alpha_4$ in Table 5-11 be reduced by a factor of 2 when applied to frequency of failure during operation." The effect of reducing theses values (and adding the difference to *1) is to reduce only the Beta factor - the gamma factors and delta factors are unchanged since the factor of one-half factors out. Contrary to this guidance, the MGL factors corresponding to the alpha factors in Table 5-11 were calculated, then the Beta factors were reduced by a factor of 2. Note these values were used in the PCC system and initiating event analyses, resulting in some factors being under-estimated by a factor of 4. The discussion in 6.3.3 regarding variable BETA1 is in error - 5 CCFs and 100 independent failures provides a beta factor of 5/105 if staggered testing is used, not the .05 indicated. A lognormal distribution is not appropriate for the GAMMA1 and DELTA1 - they should be modeled using beta distributions. | The values for GAMMA2 and DELTA2 were recalculated using the correct equations. Also beta distributions were developed for these generic distributions. With regard to the comment that BETA1 should be 5/105 rather than 0.05, these are essentially the same number.<br>- Included in 1999 PRA Update. |

| MC ID | PR | F&O ID | Peer Review Finding | Resolution |
|---|---|---|---|---|
| MC#165 | 1999 Peer Review (OG Cert) | F&O DA-6 (DA-C15) | Examine dependencies of HEPs embedded within recovery models with other human actions included in the plant model. Examine most recent component failure data to ensure recoverable failure fraction remains valid. Develop appropriate procedures for identifying and evaluating dependencies. | Operator dependencies were examined, resulting in changes made to the logic rules and HEP quantification.<br>- Included in 1999 PRA Update. |
| MC#182 | 1999 Peer Review (OG Cert) | F&O QU-3 (QU-B2) | A discussion of the limitations of using the saved sequences as a PRA model of the plant was not located. Although a very low cutoff is used to generate saved sequences, it is important that all analysts understand where limitations may exist so that they can be evaluated for specific applications | This issue of truncation has been addressed in the PRA documentation along with general guidance for setting the truncation level. Practically, this issue must be evaluated for each analysis. It is not possible to give general guidance that addresses every application.<br>- Included in 2002 PRA Update. |
| MC#185 | 1999 Peer Review (OG Cert) | F&O QU-9 (QU-E4) | At present no parametric uncertainty analysis exists based on the current plant model. While such studies were performed for earlier versions of the SSPSA, the results have significantly changed (internals are far less dominant) and the uncertainty distribution may no longer be valid. At present there is no formal analysis which addresses plant specific uncertainty or sensitivity issues. For example, cases where thermal hydraulic analyses predict only small margins for success in terms of the number of trains required, or the time available for operator actions, are prime candidates. Other examples might be cases where unique success criteria or modeling have been applied such as for feed and bleed and for RWST make up following LOCA. Perform a set of sensitivity runs and a qualitative or quantitative uncertainty analysis for the model. Risk achievement analyses may be used to focus the search for potentially significant cases. | Performed an uncertainty analysis to address this F&O. Ensured that all split fractions have an uncertainty distribution associated with them and quantified all event tree top events with Monte Carlo. Also quantified all system initiating events with Monte Carlo. Quantified uncertainty for dominant sequences for CDF and LERF.<br>- Included in 2004 PRA Update. |
| MC#198 | 1999 Peer Review (OG Cert) | F&O MU-2 (SY-A2) | During a review of plant design changes incorporated into the 1999 PRA models, it appeared that Design Change Request (DCR) 89-061 had not been incorporated into the service water fault tree. This DCR deleted the cooling tower fan auto-start feature. Therefore, a human error basic event was to be added to the service water fault tree. The service water fault tree did not appear to have been modified. Also, the PRA documentation still includes the cooling tower fans being actuated by a TA signal. It is believed that this is an isolated occurrence. However, the host utility should check for any others. Incorporate this DCR into the system fault tree / notebook. | A review of the PRA documentation (Service Water Notebook) indicated that this DCR had indeed been incorporated in the PRA model. In fact, the system notebook describes the modeling of the cooling tower and indicates that the operator must manually initiate CT operation and provides a justification for why this action is not modeled. The Service Water notebook was updated to ensure completeness. Also, a review of DCRs for the 1999 update was performed to ensure that all DCRs that impact the PRA model were addressed.<br>- Included in 1999 PRA Update. |
| MC#518 | 2005 Peer Review (AS,SC,HR) | F&O AS-A9-1 (AS-A9) | The ASME Category II capability for this SR requires the use of realistic, applicable T/H analyses for accident sequence parameters. Category III requires use of realistic, plant specific T/H analyses. Although most of the SSPSS parameters have supporting calculations that are plant specific, it appears that some would benefit from more realistic analyses. In at least one case (i.e., CST depletion) more realistic analyses may impact sequence development (and are dependent on whether the EFW pump or SUFP is running). Expectation for future applications is more extensive use of realistic codes (e.g., MAAP), as applicable. | The SSPSS-2005 update effort used MAAP to provide substantial additional plant-specific, realistic support. In some cases such as the CST example noted above, hand calculations were considered to be appropriate and were reviewed to assure adequate realism. The actions below were taken to address realistic/plant-specific success criteria: Listed all current Level 1 success criteria, including impact of power uprate, RCPs, IA, etc. Identified current basis for success criteria. Ran series of MAAP runs where needed to provide basis.<br>- Included in 2005 PRA Update. |

| MC ID | PR | F&O ID | Peer Review Finding | Resolution |
|---|---|---|---|---|
| MC#531 | 2005 Peer Review (AS,SC, HR) | F&O HR-E3-1 (HR-E3) | While simulator exercises were observed, there is no evidence of specific talk-throughs with Operations/Training. Interaction with Operations and/or Training is important regarding the assumptions used in the HRA, especially response times and performance shaping factors (PSFs), to confirm that the interpretation and implementation of the procedures are consistent with plant training and expected responses. | Walkthroughs / talk-throughs with Operations and/or Training were used to confirm modeling of operator actions and accident sequences.<br>- Included in 2009 PRA Update. |

| MC ID | PR | F&O ID | Peer Review Finding | Resolution |
|---|---|---|---|---|
| MC#536 | 2005 Peer Review (AS,SC, HR) | F&O HR-G4-1 (HR-G4) | In general, the time available to complete actions is based on either generic T/H analyses for similar Westinghouse 4-loop plants or plant-specific analyses. Several issues were identified that may point to the need for establishing a more thorough and realistic basis. For example: The write-up for the operator action ODEP1 for SBO events states that 8.8 hours are available to perform this action, which is based on 9.8 hours to core damage from WCAP-16141, less one hour to restore equipment. However, WCAP-16141 states that without depressurization, core damage can occur as early as 2.7 hours. Therefore, the time available to perform this action should not exceed the time to core damage without credit for the action. It should be noted that WCAP-16141 does not specifically mention when depressurization must begin, but it seems to be assumed that depressurization will typically begin within 30 – 45 minutes. Since this action has a low F-V and RAW importance, SR HR-G4 is judged to be satisfied. WCAP-16141, which is used as a basis, assumes that the turbine-driven AFW pump supplies 1145 gpm, which seems to exceed the capacity of the Seabrook Station TD AFWP. The basis of the time available for operator action ODEP3 does not appear to be realistic. SSPSS-2004 credits post-LOCA cooldown and depressurization for MLOCA with high head injection (HHI) success. Operator Action timing (3.8 hours) is based on a small LOCA, not MLOCA. The success criteria indicates that only 42.8 minutes are available before reaching low-low level for MLOCA. While it is true that MLOCAs at the high end of the spectrum should not require this action and MLOCAs on the low end of the spectrum behave more like a small LOCA, the majority of MLOCAs will be in between. Using the average timing between the high end (42.8 minutes) and low end (3.8 hours) would not leave enough time to successfully establish low pressure recirculation prior to reaching the RWST low-low level switchover setpoint. The time assumed to be available for feed and bleed using the Safety Injection (SI) pumps, which is based on the time until SG dryout, may not be realistic. It would seem that establishing feed and bleed with the charging pumps would have different timing than establishing feed and bleed with the SI pumps due to the lower shutoff head of the SI pumps. In particular, while waiting until SG dryout could allow successful feed and bleed cooling using the charging pumps, it isn't clear that waiting until SG dryout would allow successful feed and bleed cooling using the SI pumps. The time available for operator action HH.ORSGC2.FL is 2.3 hours, which is based on time to core damage. However, restoring secondary cooling at the time of core damage will not prevent core damage. In order to prevent core damage, secondary cooling must be completed earlier (e.g., core uncovery) With respect to the items identified: Re-evaluate the time available to perform RCS cooldown and depressurization following an SBO. Also evaluate the applicability of WCAP-14161 assumptions regarding flow from the turbine-driven AFW pump. Re-evaluate the time available used to quantify operator actions for depressurization and feed and bleed by performing sequence-specific MAAP (or other) thermal-hydraulic runs. In the case of operator action to perform depressurization for MLOCA sequences, T/H runs may need to be performed for an "average" MLOCA break size. Use MAAP or some other calculations to determine the latest time at which secondary cooling can be restored and still prevent core damage. More generally, complete the ongoing effort to establish appropriate timeframes using realistic codes (e.g. MAAP). | Revised the HRA Calculator quantification using time windows from Seabrook Station-specific MAAP runs.<br>- Included in 2005 PRA Update. |

| MC ID | PR | F&O ID | Peer Review Finding | Resolution |
|---|---|---|---|---|
| MC#538 | 2005 Peer Review (AS,SC, HR) | F&O HR-G7-1 (HR-G7) | Dependency between multiple human actions was considered, and the process for quantifying dependencies is described in SSPSS-2002. This appears to be a good approach. However, there is no guidance as to how to identify sequences with multiple operator actions for inclusion in the dependency analysis. Also, while the matrix showing dependency between two operator actions is good, it does not include new actions since the 2002 update. The review discovered at least two examples where dependencies appear to be inadequately addressed: (1) The dependency between operator actions ORSGC and OFB does not appear to be modeled, other than time consumed associated with responding to feed and bleed criteria. There is also some dependency in diagnosing the loss of secondary heat sink for these two actions. (2) The procedural guidance in Functional Restoration Procedure FR-H.1 for aligning fire water is contained in the RNO column of Step 14, which is predicated on not being able to open the PORVs. However, if the PORVs are opened too late, the procedure will not direct the operator to establish fire water to the SGs. This dependency is not modeled. Although significant progress has been made in this area since the 1999 peer review, it appears that there remains a need to develop an overall process for identifying multiple operator actions that need to be addressed in the dependency analysis. | The following actions were taken during the PRA update: 1. Identified all dynamic actions embedded in hardware top events. 2. Created new Operator Action top events, separate from hardware where appropriate. 3. For PCCW, redefined System split fractions to be conditional on Operator Action OPCC and added house events. 4. Added new top events to event trees 5. Modified logic rules to account for operator action dependency to system.<br>- Included in 2005 PRA Update. |
| MC#770 | 2009 Peer Review (IF) | F&O 4-9 (IFQU-B3) | The completeness of assumptions and sources of uncertainty in the pipe failure data (e.g., error factor, applicability of data), failure probability of doors, generic data and modeling choices needs to be reviewed against other industry studies. | A check of the data and assumptions used in the internal flooding study was performed for reasonableness and for identification of additional uncertainties. Appendix 12.1H, Uncertainties, was revised to clarify/ensure areas of uncertainty and important assumptions are adequately captured and characterized.<br>- Included in 2011 PRA Update. |

| MC ID | PR | F&O ID | Peer Review Finding | Resolution |
|---|---|---|---|---|
| MC#771 | 2009 Peer Review (IF) | F&O 5-2 (IFSO-B3, | Appendix 12.1H acknowledges uncertainty in break flow rate. Need to expand uncertainty review to discuss other source related uncertainties such as maintenance-induced events and potential, if any, source pressure or temperature impacts. Also, discuss potential for breaks or human induced events greater than assigned (i.e., catastrophic CW expansion joint failure could far exceed 56,000 gpm). Potential for larger floods can represent key insights. Specifically, CW flood rates greater than 56,000 gpm could represent a more significant threat to the Essential Switchgear rooms due to the configuration at Seabrook. | As mentioned in the disposition for F&O 4-9, Appendix 12.1H, Uncertainties, was revised to clarify/ensure areas of uncertainty and important assumptions are adequately captured and characterized. In addition, a sensitivity evaluation was performed to conservatively determine the risk significance of a postulated maximum CW flood event. The maximum CW break flow was estimated at approximately 300,000 gpm.<br>A door failure evaluation was performed to estimate the capacity of the various door configurations at Seabrook. Doors C102, C101 and C100 provide an interface between the TB and ESWGR-A. The door evaluation indicates that the capacity of these types of doors loaded against the jam/frame is in excess of any credible flood height in the TB. In addition, other doors in the Turbine Building are expected to fail at considerably less water height - approximately 10 feet (or less) and there is an unlatched door on the east side near condensate polishing that opens out. The benefit of this door was not credited. Once a flood height of ~10 ft or less is achieved, failure of these other doors (which includes the rollup doors, glass sliding door, misc. double doors) is expected to vent the flood water to outdoors and result in a steady-state water level in the TB of ~4 feet. It is noted that this TB flooding scenario is likely to cause a loss of offsite power or fail non-essential electrical buses, resulting in a trip of the flooding source – the CW pumps long before there is propagation impact in the essential switchgear rooms.<br>Based on the above, a conservative flood scenario was developed as sensitivity case F0TCWS. Based on this sensitivity case, the CDF from a postulated maximum CW break event in the TB is approximately 1E-09/yr. This scenario is screened from further detailed evaluation using criterion QN4a - Specific flood source in a flood area with CDF < ~1e-9 per yr based on flood-initiated accident sequences from a specific flood source in the flood area. This assessment is conservative. Realistic modeling would eliminate conservatisms and further reduce the impacts.<br>- Included in 2011 PRA Update. |

| MC ID | PR | F&O ID | Peer Review Finding | Resolution |
|---|---|---|---|---|
| MC#772 | 2009 Peer Review (IF) | F&O 5-3 (IFSN-A2) | The assessment indicates that there are some "rugged" doors capable of withstanding a water-height of 6-7 feet. These were walked-down for the peer review and they are indeed rugged in appearance. However, there is limited basis for door capacity other than "Industry Sources" which include a PWR OG e-mail. The EPRI Flood Guideline says the following: If there are doors within the boundaries of the area then the following guidance can be applied:<br>Water tight doors should be considered as failing only through human actions. If the door is alarmed its failure probability can be considered to be zero. If the door is not alarmed then assume the normal egress failure condition of a door opening out of the flood area if the water tight door opens out of the area. If the water tight door opens into the area then consider the failure probability to be zero.<br>Normal egress and fire doors should be considered failed after 3 foot of flood level if the door opens into the area.<br>Normal egress and fire doors should be considered failed after 1 foot of flood level if the door opens out of the flood area.<br>The 1 and 3 foot EPRI Guideline should be used unless a higher value can be justified. While the doors are clearly rugged, some more detailed justification should be presented. | A structural evaluation of typical doors at Seabrook Station was performed and documented in a calculation, "Structural Evaluation of Door Capacity Under Flooding Loading Conditions". The evaluation was performed for 3 "typical"-type doors including: (1) rugged security door, (2) industrial 3 hour rated fire door, and (3) double-wide industrial door with and without a center locking pin. The evaluation addressed the difference in potential failure when each type of door is loaded against its frame/jamb (stronger door configuration) verses being loaded against its latch and hinges (weaker door configuration). It is noted that the door frames at Seabrook are embedded into the adjacent concrete and are not supported by installed anchor bolts. This represents a much stronger configuration than a conventionally installed frame with anchor bolts.<br>Door capacity/failure insights from the structural evaluation are included in Appendix 12.1A, Methodology. Door failures and the resultant propagation are assessed on an individual door/scenario basis. If the scenario's flood water height does not exceed the door's capacity, the door is not expected to fail, is assumed to remain intact with only gap leakage contributing to propagation. On the contrary, if the scenario's flood water height exceeds the door capacity, door failure is assumed and the resulting propagation is via the failed (open) door. No credit is given for failure of a barrier to limit the flood consequence without some assessment of the door failure potential.<br>- Included in 2011 PRA Update. |

## *Attachment B Supporting Requirements That Meet Less Than CC II*

Section 3.1 identified that only six of the 254 supporting requirements in Part 2 were judged to not meet at least Capability Category II. These six SRs, listed below, were assessed in the 2012 Focused Peer Review for element LE. This table documents the open peer review F&O supporting this assessment and the significance evaluation performed for each open item following the peer review.

| SR # | Supporting Requirement | 2012 PR Assessment | Basis for PR Assessment (Related F&O) | Significance Evaluation |
|---|---|---|---|---|
| LE-C3 | CC II,III: REVIEW significant accident progression sequences resulting in a large early release to determine if repair of equipment can be credited. JUSTIFY credit given for repair [i.e., ensure that plant conditions do not preclude repair and actuarial data exists from which to estimate the repair failure probability (see SY-A24, DA-C15, and DA-D8)]. AC power recovery based on generic data applicable to the plant is acceptable. | Met CC I | No credit for repair was taken in the analysis other than recovery of AC power. There was no review of the accident progression sequences for opportunities to credit equipment repair.<br>F&O: LE-C3-01 | MCDB #880 (Significance "C")<br>This finding is primarily a documentation issue to provide some basis for why repair would not be credited for LERF sequences. By definition, the time available for repair for LERF sequences is short and it would be difficult to provide any credit for repair beyond offsite power. |
| LE-C5 | CC II: USE appropriate realistic generic or plant-specific analyses for system success criteria for the significant accident progression sequences. USE conservative or a combination of conservative and realistic system success criteria for non-risk significant accident progression sequences. | Met CC I | The only relevant system looked at for this SR was AFW (for SGTR scrubbing). No basis for the AFW success criteria, as documented in the SSPSS, Section10.4.3.3, is given. It is assumed that these success criteria are based on design calculations, not realistic analyses.<br>F&O: LE-C5-01 | MCDB #881 (Significance "C")<br>This finding is primarily a documentation issue to provide some basis for crediting of AFW for SGTR scrubbing. Justification is based on generic (PWROG) analyses developed in support of SAMGs. |
| LE-C10 | CC II: REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions during accident progression that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences. | Met CC I | Although no credit is given to equipment and operator response in an adverse environment (probably no equipment in these environments), the SSPSS updates in WCAP-16600 considered new models including SG scrubbing and operator responses from SAMG that could reduce LERF. However, there is no evidence of the REVIEW.<br>F&O: LE-C10-01 | MCDB #883 (Significance "C")<br>This finding is a review and documentation issue to provide a basis for not crediting any additional equipment or operator actions to reduce LERF. In general SAMG actions and associated equipment were credited where the actions were feasible and equipment available and where the credit would change the classification of the release category for a sequence. Thus, for example, while FP spray of a release may reduce the release somewhat, it would be difficult to provide a basis to justify sufficient reduction to move such a sequence from LERF. |

| SR # | Supporting Requirement | 2012 PR Assessment | Basis for PR Assessment (Related F&O) | Significance Evaluation |
|---|---|---|---|---|
| LE-C12 | CC II: REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions after containment failure that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences. | Met CC I | Although no credit is given to equipment and operator response post containment failure (probably irrelevant and too late), the SSPSS updates in WCAP-16600 considered new models including SG scrubbing and operator responses from SAMG that could reduce LERF. However, there is no evidence of the REVIEW.<br>F&O: LE-C10-01 | MCDB #883 (Significance "C")<br>This finding is the same as the F&O for LE-C10 since LE-C10 and LE-C12 are closely related. LE-C10 addresses "during accident progression" while LE-C12 addresses "after containment failure." The same basis for "significance evaluation" applies for both LE-C10 and LE-C12. |
| LE-D6 | CC II: PERFORM an analysis of thermally-induced SG tube rupture that includes plant-specific procedures and design features and conditions that could impact tube failure. An acceptable approach is one that arrives at plant-specific split fractions by selecting the SG tube conditional failure probabilities based on NUREG-1570 or similar evaluation for induced SG failure of similarly designed SGs and loop piping. SELECT failure probabilities based on (a) RCS and SG post-accident conditions sufficient to describe the important risk outcomes (b) secondary side conditions including plant-specific treatment of MSSV and ADV failures. JUSTIFY assumptions and selection of key inputs. An acceptable justification can be obtained by the extrapolation of the information in NUREG-1570 to obtain plant-specific models, use of reasonably bounding assumptions, or performance of sensitivity studies indicating low sensitivity to changes in the range in question. | Not Met | Because Seabrook has relatively young SGs, the likelihood of thermally induced tube rupture is low. Seabrook uses a probability of 1E-3, which also follows a probability of 1E-2 that the hot leg piping does not rupture. However, the analysis does not consider an increased probability due to depressurized steam generators that may occur due to secondary side conditions as mentioned in item (b). In addition, because thermally-induced tube rupture follows hot leg integrity in the event tree, proper consideration of the conditional probabilities should be re-addressed.<br>F&O: LE-D6-01 | MCDB #884 (Significance "C")<br>This finding identifies two issues, both of which were evaluated to be minor issues, requiring improved documentation.<br><br>ISSUE (1): The Seabrook analysis does not consider an increased probability due to depressurized steam generators that may occur due to secondary side conditions as mentioned in item (b) of LE-D6.<br>EVALUATION: In fact, the SBK PRA (Section 14.3.3.1) documents the failure probabilities for induced SGTR given secondary side depressurization due to failure of MSSVs (see split fraction SGTI2). MSSV failure would clearly bound the impact of an ASDV failure.<br><br>ISSUE (2): Because thermally-induced tube rupture follows hot leg integrity in the event tree, proper consideration of the conditional probabilities should be re-addressed.<br>EVALUATION:<br>Thermally-induced SG tube rupture that occurs prior to core melt (e.g., induced tube rupture due to steam line break) is modeled in top event SGTI. This includes conditional tube rupture probabilities as high as 0.1 (see Section 14.3.3.1). In addition, thermally-induced SG tube rupture that occurs following core melt is modeled in top event XSGTI. This failure mode has low probabilities due to the strength of the SG tubes and the more likely failure of hot leg creep rupture. However, even two orders of magnitude increase in this probability (1e-3 to 0.1) would increase LERF less than 1%. |

| SR # | Supporting Requirement | 2012 PR Assessment | Basis for PR Assessment (Related F&O) | Significance Evaluation |
|---|---|---|---|---|
| LE-G6 | CC I,II,III: DOCUMENT the quantitative definition used for significant accident progression sequence. If other than the definition used in Section 2, JUSTIFY the alternative. | Not Met | No documentation of the quantitative definition used for a significant accident progression sequence was found.<br>F&O: LE-G6-01 | MCDB #887 (Significance "D")<br>This finding is strictly a documentation only issue since the definition of "significant accident progression sequence" was used in the LE evaluation in the Seabrook PRA. |

# Attachment 3

Mark-up of Proposed Technical Specification Changes

**Mark-up of Proposed Technical Specification Changes**

The attached markups reflect the currently issued version of the TS and Facility Operating License. At the time of submittal, the Facility Operating License was revised through Amendment No. 136.

Listed below are the license amendment requests that are awaiting NRC approval and may impact the currently issued version of the Facility Operating License affected by this LAR.

| LAR | Title | NextEra Energy Seabrook Letter | Date Submitted |
|---|---|---|---|
| 11-04 | Changes to the Technical Specifications for New and Spent Fuel Storage | SBK-L-11245 | 01/30/2012 |
| 13-01 | Application for Administrative Change and Corrections to the Technical Specifications | SBK-L-13028 | 03/01/2013 |
| 13-02 | Application to Revise Technical Specifications to Adopt TSTF-510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," Using the Consolidated Line Item Improvement Process | SBK-L-13030 | 03/27/2013 |

**INSERT 1**

In accordance with the Surveillance Frequency Control Program

**INSERT 2**

n. 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.

# INDEX

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 1.1

FREQUENCY NOTATION

| NOTATION | FREQUENCY |
|---|---|
| S | At least once per 12 hours. |
| D | At least once per 24 hours. |
| W | At least once per 7 days. |
| M | At least once per 31 days. |
| Q | At least once per 92 days. |
| SA | At least once per 184 days. |
| R | At least once per 18 months. |
| S/U | Prior to each reactor startup. |
| N.A. | Not applicable. |
| P | Completed prior to each release. |
| SFCP | In accordance with the Surveillance Frequency Control Program |

TABLE 1.2

OPERATIONAL MODES

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

*Excluding decay heat.

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

## 3/4.1 REACTIVITY CONTROL SYSTEMS

### 3/4.1.1 BORATION CONTROL

### SHUTDOWN MARGIN -$T_{avg}$ GREATER THAN 200°F

### LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN for four-loop operation shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

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

ACTION:

With the SHUTDOWN MARGIN less than the limiting value, immediately initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System 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 the limiting value:

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

b. When in MODE 1 or MODE 2 with $k_{eff}$ greater than or equal to 1 ~~at least once per 12 hours~~ INSERT 1 by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with $k_{eff}$ less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

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

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 (Continued)

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

INSERT 1

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

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

SHUTDOWN MARGIN - $T_{avg}$ LESS THAN OR EQUAL TO 200°F

## LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR). Additionally, the Reactor Coolant System boron concentration shall be greater than or equal to the limit specified in the COLR when the reactor coolant loops are in a drained condition.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than the limit specified in the COLR or the Reactor Coolant System boron concentration less than the limit specified in the COLR, immediately initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the required SHUTDOWN MARGIN and boron concentration are restored.

## SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limit specified in the COLR and the Reactor Coolant System boron concentration shall be determined to be greater than or equal to the limit specified in the COLR when the reactor coolant loops are in a drained condition:

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 verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

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

INSERT 1

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

### ISOLATION OF UNBORATED WATER SOURCES - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.7 Provisions to isolate the Reactor Coolant System from unborated water sources shall be OPERABLE with:

a. The Boron Thermal Regeneration System (BTRS) isolated from the Reactor Coolant System, and

b. The Reactor Makeup Systems inoperable except for the capability of delivering up to the capacity of one Reactor Makeup Water pump to the Reactor Coolant System.

APPLICABILITY: MODES 4, 5, and 6

ACTION:

With the requirements of the above specification not satisfied immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and, if within 1 hour the required SHUTDOWN MARGIN is not verified, initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limits specified in the COLR for the Boric Acid Storage System until the required SHUTDOWN MARGIN is restored and the isolation provisions are restored to OPERABLE. 4

### SURVEILLANCE REQUIREMENTS

4.1.2.7 The provisions to isolate the Reactor Coolant System from unborated water sources shall be determined to be OPERABLE ~~at least once per 31 days~~ by: INSERT 1

a. Verifying that at least the BTRS outlet valve, CS-V-302, or the BTRS moderating heat exchanger outlet valve, CS-V-305, or the manual outlet isolation valve for each demineralizer* not saturated with boron, CS-V-284, CS-V-295, CS-V-288, CS-V-290, CS-V-291, is closed and locked closed, and

b. Verifying that power is removed from at least one of the Reactor Makeup Water pumps, RMW-P-16A or RMW-P-16B.

*A demineralizer may be unisolated to saturate a bed with boron provided the effluent is not directed back to the Reactor Coolant System.

## LIMITING CONDITION FOR OPERATION

3.1.3.1 ACTION b.3 (Continued)

c) A power distribution map is obtained from the Incore Detector System and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

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

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a. above, POWER OPERATION may continue provided that:

1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

d. With more than one rod misaligned from its group step counter demand height by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

## SURVEILLANCE REQUIREMENTS

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

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 92 days~~. A

## REACTIVITY CONTROL SYSTEMS

## MOVABLE CONTROL ASSEMBLIES

## POSITION INDICATION SYSTEMS - OPERATING

### LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

   1. Determine the position of the nonindicating rod(s) indirectly by the Incore Detector System at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or (4)

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

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

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

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

### SURVEILLANCE REQUIREMENTS

INSERT 1

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

## POSITION INDICATION SYSTEM - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

### SURVEILLANCE REQUIREMENTS

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

INSERT 1

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

**See Special Test Exceptions Specification 3.10.5.

REACTIVITY CONTROL SYSTEMS

MOVABLE CONTROL ASSEMBLIES

ROD DROP TIME

## LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the mechanical fully withdrawn position shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with: 1

a. $T_{avg}$ for each loop greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

## SURVEILLANCE REQUIREMENTS

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

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

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

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

INSERT 1

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn# as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod not fully withdrawn#, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn# as specified in the COLR:

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

b. ~~At least once per 12 hours~~ thereafter.

INSERT 1

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.
**With $k_{eff}$ greater than or equal to 1.
#The fully withdrawn position is defined as the interval within 225 to the mechanical fully withdrawn position, inclusive.

REACTIVITY CONTROL SYSTEMS

MOVABLE CONTROL ASSEMBLIES

CONTROL ROD INSERTION LIMITS

## LIMITING CONDITION FOR OPERATION

3.1.3.6 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 insertion limits specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2:

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

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

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

## SURVEILLANCE REQUIREMENTS

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

INSERT 1

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4.2.1 AXIAL FLUX DIFFERENCE

### SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel ~~at least once per 7 days~~ INSERT 1 when the AFD Monitor Alarm is OPERABLE, and

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

4.2.1.2 (THIS SPECIFICATION NUMBER IS NOT USED)

## SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits by:

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

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% when using the moveable incore detectors or 5.21% when using the fixed incore detectors to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(Z) \le \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \quad \text{for } P > 0.5$$

$$F_Q^M(Z) \le \frac{F_Q^{RTP} \times K(Z)}{0.5 \times W(Z)} \quad \text{for } P \le 0.5$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, $F_Q^{RTP}$ is the $F_Q$ limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the relative THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. $F_Q^{RTP}$, K(Z), and W(Z) are specified in the COLR.

d. Measuring $F_Q^M(Z)$ according to the following schedule:

1) Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined*, or

2) ~~At least once per 31 Effective Full Power Days (EFPD)~~, whichever occurs first.

INSERT 1

* During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

## LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

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

a. Within 2 hours reduce the THERMAL POWER to the level where the LIMITING CONDITION FOR OPERATION is satisfied.

b. THERMAL POWER may be increased, provided $F^N_{\Delta H}$ is demonstrated through incore mapping to be within its limit.

## SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

INSERT 1

4.2.3.2 $F^N_{\Delta H}$ shall be demonstrated to be within its limit prior to operation above 75% RATED THERMAL POWER after each fuel loading and ~~at least once per 31 EFPD~~ thereafter by:

a. Using the Incore Detector System to obtain a power distribution map at any THERMAL POWER greater than 5% RATED THERMAL POWER.

b. Using the measured value of $F^N_{\Delta H}$ which does not include an allowance for measurement uncertainty.

## 3/4.2.4 QUADRANT POWER TILT RATIO

### LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

With the QUADRANT POWER TILT RATIO determined to exceed 1.02:

a. Within 2 hours reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

b. Within 24 hours and every 7 days thereafter, verify that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits by performing Surveillance Requirements 4.2.2.2 and 4.2.3.2. THERMAL POWER and setpoint reductions shall then be in accordance with the ACTION statements of Specifications 3.2.2 and 3.2.3.

### 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, and

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the Incore Detector System to confirm indicated QUADRANT POWER TILT RATIO ~~at least once per 12 hours~~ by either:

INSERT 1

a. Using the four pairs of symmetric detector locations or

b. Using the Incore Detector System to monitor the QUADRANT POWER TILT RATIO subject to the requirements of Technical Requirement TR20-3.3.3.2.

*See Special Test Exceptions Specification 3.10.2

## 3/4.2.5 DNB PARAMETERS

### LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System $T_{avg}$ is less than or equal to the limit specified in the COLR,

b. Pressurizer Pressure is greater than or equal to the limit specified in the COLR*, and

c. Reactor Coolant System Flow shall be:

1. ≥ 374,400 gpm**; and,

2. ≥ 383,800 gpm***

APPLICABILITY: MODE 1.

ACTION:

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

### SURVEILLANCE REQUIREMENTS

INSERT 1

4.2.5.1 Each of the parameters shown above shall be verified to be within its limits ~~at least once per 12 hours~~.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION ~~at least once per 18 months~~.

4.2.5.3 The RCS total flow rate shall be determined by an approved method to be within its limit prior to operation above 95% of RATED THERMAL POWER after each fuel loading. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1.

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

**Thermal Design Flow. An allowance for measurement uncertainty shall be made when comparing measured flow to Thermal Design Flow.

***Minimum measured flow used in the Revised Thermal Design Procedure.

## 3/4.3 INSTRUMENTATION

## 3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

### SURVEILLANCE REQUIREMENTS

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

INSERT 1

4.3.1.2 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~~. Each verification shall include at least one train ~~such that both 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~~.

Replace each marked through surveillance frequency with: SFCP

## TABLE 4.3-1

## REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|
| 1. Manual Reactor Trip | N.A. | N.A. | N.A. | ~~R~~(13) | N.A. | 1,2,3*,4*,5* |
| 2. Power Range, Neutron Flux | | | | | | |
| a. High Setpoint | ~~S~~ | ~~D~~(2, 4), ~~M~~(3, 4), ~~Q~~(4, 6), ~~R~~(4, 5) | ~~Q~~ | N.A. | N.A. | 1, 2 |
| b. Low Setpoint | ~~S~~ | ~~R~~(4) | S/U(1) | N.A. | N.A. | 1***, 2 |
| 3. Power Range, Neutron Flux, High Positive Rate | N.A. | ~~R~~(4) | ~~Q~~ | N.A. | N.A. | 1, 2 |
| 4. (NOT USED) | | | | | | |
| 5. Intermediate Range, Neutron Flux | ~~S~~ | ~~R~~(4, 5) | S/U(1) | N.A. | N.A. | 1***, 2 |
| 6. Source Range, Neutron Flux | ~~S~~ | ~~R~~(4, 5) | S/U(8), ~~Q~~(9) | N.A. | N.A. | 2**, 3, 4, 5 |
| 7. Overtemperature ΔT | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | 1, 2 |
| 8. Overpower ΔT | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | 1, 2 |
| 9. Pressurizer Pressure--Low | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | 1 |
| 10. Pressurizer Pressure--High | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | 1, 2 |
| 11. Pressurizer Water Level--High | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | 1 |
| 12. Reactor Coolant Flow--Low | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | 1 |

Replace each marked through surveillance frequency with: SFCP

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|
| 13. Steam Generator Water Level--Low-Low | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | 1, 2 |
| 14. Undervoltage - Reactor Coolant Pumps | N.A. | ~~R~~ | N.A. | ~~Q~~ | N.A. | 1 |
| 15. Underfrequency - Reactor Coolant Pumps | N.A. | ~~R~~ | N.A. | ~~Q~~ | N.A. | 1 |
| 16. Turbine Trip | | | | | | |
| a. Low Fluid Oil Pressure | N.A. | ~~R~~ | N.A. | S/U(8, 10) | N.A. | 1 |
| b. Turbine Stop Valve | N.A. | ~~R~~ | N.A. | S/U(8, 10) | N.A. | 1 |
| 17. Safety Injection Input from ESF | N.A. | N.A. | N.A. | ~~R~~ | N.A. | 1, 2 |
| 18. Reactor Trip System Interlocks | | | | | | |
| a. Intermediate Range Neutron Flux, P-6 | N.A. | ~~R~~(4) | ~~R~~ | N.A. | N.A. | 2** |
| b. Low Power Reactor Trips Block, P-7 | N.A. | ~~R~~(4) | ~~R~~ | N.A. | N.A. | 1 |
| c. Power Range Neutron Flux, P-8 | N.A. | ~~R~~(4) | ~~R~~ | N.A. | N.A. | 1 |
| d. Power Range Neutron Flux, P-9 | N.A. | ~~R~~(4) | ~~R~~ | N.A. | N.A. | 1 |

Replace each marked through surveillance frequency with: SFCP

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|
| Reactor Trip System Interlocks (Continued) | | | | | | |
| e. Power Range Neutron Flux, P-10 | N.A. | ~~R~~(4) | ~~R~~ | N.A. | N.A. | 1, 2 |
| f. Turbine Impulse Chamber Pressure, P-13 | N.A. | ~~R~~ | ~~R~~ | N.A. | N.A. | 1 |
| 19. Reactor Trip Breaker | N.A. | N.A. | N.A. | ~~M~~(7, 11) | N.A. | 1, 2, 3*, 4*, 5* |
| 20. Automatic Trip and Interlock Logic | N.A. | N.A. | N.A. | N.A. | ~~M~~(7) | 1, 2, 3*, 4*, 5* |
| 21. Reactor Trip Bypass Breaker | N.A. | N.A. | N.A. | ~~M~~(7, 14), ~~R~~(15) | N.A. | 1, 2, 3*, 4*, 5* |

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be closed and the Control Rod Drive System is capable of rod withdrawal.

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

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

(1) If not performed in previous 92 days.

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

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 50% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purposes of this surveillance requirement, monthly shall mean at least once per 31 EFPD.

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

(5) Initial plateau curves shall be measured for each detector. Subsequent plateau curves shall be obtained, evaluated and compared to the initial curves. The plateau curves for the Intermediate Range and Power Range detectors are required to be measured or obtained within 24 hours after attaining 100% of RATED THERMAL POWER. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purposes of this surveillance requirement, quarterly shall mean at least once per 92 EFPD.

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

INSERT 1

(8) If not performed in previous 31 days.

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

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

## SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2. INSERT 1

4.3.2.2 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 train ~~such that both 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 ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.~~

Replace each marked through surveillance frequency with: SFCP

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generator, Phase "A" Isolation, Containment Ventilation Isolation, and Emergency Feedwater, Service Water to Secondary Component Cooling Water Isolation, CBA Emergency Fan/Filter Actuation, and Latching Relay). | | | | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | ~~R~~ | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | ~~M~~(1) | ~~M~~(1) | ~~Q~~ | 1, 2, 3, 4 |
| c. Containment Pressure-Hi-1 | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| d. Pressurizer Pressure Low | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| e. Steam Line Pressure-Low | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| 2. Containment Spray | | | | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | ~~R~~ | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | ~~M~~(1) | ~~M~~(1) | ~~Q~~ | 1, 2, 3, 4 |
| C. Containment Pressure-Hi-3 | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |

Replace each marked through surveillance frequency with: SFCP

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

Replace each marked through surveillance frequency with: SFCP

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 4. Steam Line Isolation | | | | | | | | |
| a. Manual Initiation (System) | N.A. | N.A. | N.A. | ~~R~~ | N.A. | N.A. | N.A. | 1, 2, 3 |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | ~~M~~(1) | ~~M~~(1) | ~~Q~~ | 1, 2, 3 |
| c. Containment Pressure-Hi-2 | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| d. Steam Line Pressure-Low | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| e. Steam Line Pressure-Negative Rate-High | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | N.A. | N.A. | 3 |
| 5. Turbine Trip | | | | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | ~~M~~(1) | ~~M~~(1) | ~~Q~~ | 1, 2 |
| b. Steam Generator Water Level-High-High (P-14) | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | N.A. | N.A. | 1, 2 |
| 6. Feedwater Isolation | | | | | | | | |
| a. Steam Generator Water Level--High-High (P-14) | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | N.A. | N.A. | 1, 2 |
| b. Safety Injection | See Item 1. above for all Safety Injection Surveillance Requirements. | | | | | | | |
| 7. Emergency Feedwater | | | | | | | | |
| a. Manual Initiation | | | | | | | | |
| 1) Motor-driven pump | N.A. | N.A. | N.A. | ~~R~~ | N.A. | N.A. | N.A. | 1, 2, 3 |
| 2) Turbine-driven pump | N.A. | N.A. | N.A. | ~~R~~ | N.A. | N.A. | N.A. | 1, 2, 3 |

Replace each marked through surveillance frequency with: SFCP

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 7. Emergency Feedwater (Continued) | | | | | | | | |
| b. Automatic Actuation and Actuation Relays | N.A. | N.A. | N.A. | N.A. | ~~M~~(1) | ~~M~~(1) | ~~Q~~ | 1, 2, 3 |
| c. Steam Generator Water Level-Low-Low, Start Motor-Driven Pump and Turbine-Driven Pump | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| d. Safety Injection, Start Motor-Driven Pump and Turbine-Driven Pump | See Item 1. above for all Safety Injection Surveillance Requirements. | | | | | | | |
| e. Loss-of-Offsite Power Start Motor-Driven Pump and Turbine-Driven Pump | See Item 9. for all Loss-of-Offsite Power Surveillance Requirements. | | | | | | | |
| 8. Automatic Switchover to Containment Sump | | | | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | ~~M~~(1) | ~~M~~(1) | ~~Q~~ | 1, 2, 3, 4 |
| b. RWST Level Low-Low Coincident With | N.A. | ~~R~~ | ~~Q~~ | ~~Q~~(3) | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| Safety Injection | See Item 1. above for all Safety Injection Surveillance Requirements. | | | | | | | |

Replace each marked through surveillance frequency with: SFCP

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| CHANNEL FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MASTER RELAY TEST | SLAVE RELAY TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|---|---|
| 9. Loss of Power (Start) Emergency Feedwater) | | | | | | | | |
| a. 4.16 kV Bus E5 and E6 Loss of Voltage | N.A. | ~~R~~ | N.A. | ~~M~~ | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| b. 4.16 kV Bus E5 and E6 Degraded Voltage | N.A. | ~~R~~ | N.A. | ~~M~~ | N.A. | N.A. | N.A. | 1, 2, 3, 4 |
| Coincident With Safety Injection | See Item 1. above for all Safety Injection Surveillance Requirements | | | | | | | |
| 10. Engineered Safety Features Actuation System Interlocks | | | | | | | | |
| a. Pressurizer Pressure, P-11 | N.A. | ~~R~~ | ~~Q~~ | N.A. | N.A. | N.A. | N.A. | 1, 2, 3 |
| b. Reactor Trip, P-4 | N.A. | N.A. | N.A. | N.A. | ~~R~~ | N.A. | N.A. | 1, 2, 3 |
| c. Steam Generator Water Level, P-14 | ~~S~~ | ~~R~~ | ~~Q~~ | N.A. | ~~M~~(1) | ~~M~~(1) | ~~Q~~ | 1, 2, 3 |

TABLE NOTATION

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

(2) A DIGITAL CHANNEL OPERATIONAL TEST will be performed on this instrumentation.

(3) Setpoint verification is not applicable.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

## LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

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

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

c. The provisions of Specification 3.0.3 are not applicable.

## SURVEILLANCE REQUIREMENTS

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

INSERT 1

TABLE 3.3-6
RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

| FUNCTIONAL UNIT | CHANNELS TO TRIP/ALARM | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ALARM/TRIP SETPOINT | ACTION |
|---|---|---|---|---|---|
| 1. Containment | | | | | |
| a. Containment - Post LOCA - Area Monitor | 1 | 2 | All | ≤ 10 R/h | 27 |
| 2. Containment Ventilation Isolation | | | | | |
| a. On Line Purge Monitor | 1 | 2 | 1, 2, 3, 4 | * | 23 |
| b. Manipulator Crane Area Monitor | 1 | 2 | 6# | ** | 23 |
| 3. Main Steam Line | 1/steam line | 1/steam line | 1, 2, 3, 4 | N.A. | 27 |
| 4. Fuel Storage Pool Areas | | | | | |
| a. Fuel Storage Building Exhaust Monitor | N.A. | 1 | *** | **** | 25 |
| 5. Control Room Isolation | | | | | |
| a. Air Intake-Radiation Level | | | | | |
| 1) East Air Intake | 1/intake | 2/intake | All | **** | 24 |
| 2) West Air Intake | 1/intake | 2/intake | All | **** | 24 |
| 6. Primary Component Cooling Water | | | | | |
| a. Loop A | 1 | 1 | All | ≤ 2 x Background | 28 |
| b. Loop B | 1 | 1 | All | ≤ 2 x Background | 28 |

No Changes on this page
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TABLE NOTATIONS

* Two times background; purge rate will be verified to ensure compliance with ODCM Control C.7.1.1 requirements

** Two times background or 15 mR/hr, whichever is greater.

*** With irradiated fuel in the fuel storage pool areas.

**** Two times background or 100 CPM, whichever is greater.

\# During CORE ALTERATIONS or movements of irradiated fuel within the containment.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | DIGITAL CHANNEL OPERATIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|
| 1. Containment | | | | |
| a. Containment - Post LOCA - Area Monitor | S | R | Q | All |
| 2. Containment Ventilation Isolation | | | | |
| a. On Line Purge Monitor | S | R | Q | 1, 2, 3, 4 |
| b. Manipulator Crane Area Monitor | S | R | Q | 6# |
| 3. Main Steam Line | S | R | Q | 1, 2, 3, 4 |
| 4. Fuel Storage Pool Areas | | | | |
| a. Radioactivity-High-Gaseous Radioactivity | S | R | Q | * |
| 5. Control Room Isolation | | | | |
| a. Air Intake Radiation Level | | | | |
| 1) East Air Intake | S | R | Q | All |
| 2) West Air Intake | S | R | Q | All |
| 6. Primary Component Cooling Water | | | | |
| a. Loop A | S | R | Q | All |
| b. Loop B | S | R | Q | All |

TABLE NOTATIONS

* With irradiated fuel in the fuel storage pool areas.
# During CORE ALTERATIONS or movement of irradiated fuel within the containment.

INSTRUMENTATION

MONITORING INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

## LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown System transfer switches, power, controls and monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

b. With the number of OPERABLE remote shutdown monitoring channels less than the Total Number of Channels as required by Table 3.3-9, within 60 days restore the inoperable channel(s) to OPERABLE status or, pursuant to Specification 6.8.2, submit a Special Report that defines the corrective action to be taken.

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

## SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel in Table 3.3-9 shall be demonstrated OPERABLE:

INSERT 1

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

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

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit listed in Table 3.3-9, including the actuated components, shall be demonstrated OPERABLE ~~at least once per 18 months~~.

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE:

a. ~~Every 31 days~~ by performance of a CHANNEL CHECK, and
b. ~~Every 18 months~~ by performance of a CHANNEL CALIBRATION.

INSERT 1

TABLE 4.3-6

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | SOURCE CHECK | CHANNEL CALIBRATION | CHANNEL OPERATIONAL TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|
| 1. RADIOACTIVE GAS WASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM | | | | | |
| Oxygen Monitor (Process) | ~~D~~ SFCP | N.A. | ~~Q~~(4) SFCP | ~~M~~ SFCP | ** |

## 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 reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

### SURVEILLANCE REQUIREMENTS

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

INSERT 1

REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

HOT STANDBY

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, shall be determined OPERABLE ~~once per 7 days~~ by verifying correct breaker alignments and indicated power availability*.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 14% ~~at least once per 12 hours~~.

INSERT 1

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

*Not required to be performed until 24 hours after a required pump is not in operation.

REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

HOT SHUTDOWN

## SURVEILLANCE REQUIREMENTS

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

INSERT 1

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary-side water level to be greater than or equal to 14% ~~at least once per 12 hours~~.

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

REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

COLD SHUTDOWN - LOOPS FILLED

## LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The secondary-side water level of at least two steam generators shall be greater than 14%.

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

ACTION:

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

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at ~~least~~ once ~~per~~ 12 ~~hours~~.

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

INSERT 1

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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

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

REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

COLD SHUTDOWN - LOOPS NOT FILLED

## LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

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

4.4.1.4.2 At least one RHR loop shall be determined 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 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 92% of pressurizer level (1656 cubic feet), and at least two groups of pressurizer heaters each having a capacity of at least 150 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, fully insert all rods, place the Control Rod Drive System in a condition incapable of rod withdrawal, and be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters from the emergency power supply and measuring circuit current ~~at least once each refueling interval~~

INSERT 1

SURVEILLANCE REQUIREMENTS

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

INSERT 1

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel during MODES 3 or 4.

4.4.4.2 Each block valve shall be demonstrated OPERABLE ~~at least once per 92 days~~ by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

## 3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

## LEAKAGE DETECTION SYSTEMS

### LIMITING CONDITION FOR OPERATION

3. Restore either the containment drainage sump level monitoring system or the containment atmosphere particulate monitor to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Required containment atmosphere radioactivity monitor:

1. Performance of a CHANNEL CHECK ~~at least once per 12 hours~~,

2. Performance of a DIGITAL CHANNEL OPERATIONAL TEST ~~at least once per 92 days~~, and

3. Performance of a CHANNEL CALIBRATION ~~at least once per 18 months~~.

b. Containment Drainage Sump Level Monitoring System - performance of CHANNEL CALIBRATION ~~at least once per 18 months~~.

INSERT 1

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

3.4.6.2

ACTION: (Continued)

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

4.4.6.2.1 Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:

a. Not Used

b. Not Used

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig ~~at least~~ ~~once per~~ 31 ~~days~~ with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

INSERT 1

d. Performance of a Reactor Coolant System water inventory balance ~~at least~~ ~~once per~~ 72 ~~hours~~ during steady-state operation, except that not more than 96 hours shall elapse between any two successive inventory balances; (1) (2)

e. Monitoring the Reactor Head Flange Leakoff System at ~~least~~ ~~once per~~ 2~~4 hours~~, and

f. Verifying primary to secondary leakage is ≤ 150 gallons per day through any one SG ~~at least once~~ per 7~~2 hours~~. (2)

---

(1) Not applicable to primary to secondary leakage.

(2) Not required to be performed until 12 hours after establishment of steady state operation.

SURVEILLANCE REQUIREMENTS

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation 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 7 days 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, and

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.*

e. Testing pursuant to Specification 4.0.5.

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

* Not applicable to RHR Pumps 8A and 8B suction isolation valves.

TABLE 4.4-3

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS ~~PROGRAM~~

| TYPE OF MEASUREMENT AND ANALYSIS | SAMPLE AND ANALYSIS FREQUENCY | MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED |
|---|---|---|
| 1. Gross Radioactivity Determination | At ~~least once per 72 hours~~. | 1, 2, 3, 4 |
| 2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration | 1 ~~per 14 days~~ | 1 |
| 3. Radiochemical for Ē Determination* | 1 ~~per 6 months~~** | 1 |
| 4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135 | a) Once per 4 hours, whenever the specific activity exceeds 1 μCi/gram DOSE EQUIVALENT 1-131 or 100/Ē Ci/gram of gross radioactivity, and | 1#, 2#, 3#, 4#, 5# |
| | b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period. | 1, 2, 3 |

INSERT 1

* A radiochemical analysis for Ē shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of Ē for the reactor coolant sample. Determination of the contributors to Ē shall be based upon those energy peaks identifiable with a 95% confidence level.

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

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

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

GENERAL

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

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

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS $T_{avg}$ and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIRMENTS

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

INSERT 1

REACTOR COOLANT SYSTEM

PRESSURE/TEMPERATURE LIMITS

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 1-hour period,

b. A maximum cooldown of 200°F in any 1-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

4.4.9.2 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

LIMITING CONDITION FOR OPERATION

3.4.9.3

ACTION: (Continued)

f) With more than one charging pump capable of injecting into the RCS, immediately initiate action to restore a maximum of one charging pump capable of injecting into the RCS.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE when the PORV(s) are being used for overpressure protection by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at ~~least once per~~ ~~31 days~~ thereafter when the PORV is required OPERABLE; and

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at ~~least once per~~ 18 ~~months~~; and

c. Verifying the PORV isolation valve is open at ~~least~~ once ~~per~~ 72 ~~hours~~.

INSERT 1

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

a. For RHR suction relief valve RC-V89 by verifying at least ~~once per 72 hours~~ that RHR suction isolation valves RC-V87 and RC-V88 are open.

b. For RHR suction relief valve RC-V24 by verifying at ~~least once per 72 hours~~ that RHR suction isolation valves RC-V22 and RC-V23 are open.

c. Testing pursuant to Specification 4.0.5.

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

---

**Except when the vent pathway is provided with a valve(s) or device(s) that is locked, sealed, or otherwise secured in the open position, then verify this valve(s) or device(s) open at ~~least~~ once per ~~31 days~~.

4.4.9.3.4 The reactor vessel water level shall be verified to be lower than 36 inches below the reactor vessel flange ~~at least once per 12 hours~~ INSERT 1 when the reduced inventory condition is being used for overpressure protection.

4.4.9.3.5 All charging pumps, excluding one OPERABLE pump, shall be demonstrated inoperable*** by verifying that the motor circuit breakers are secured in the open position**** ~~at least once per 31 days~~ INSERT 1, except when the reactor vessel head closure bolts are fully detensioned or the vessel head is removed.

*** An additional pump may be made capable of injecting under administrative control for up to 1 hour during pump-swap operation, except during RCS water-solid conditions. Additionally, an inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

**** An alternate method to assure pump inoperability may be used by placing the control room pump-control switch in the Pull-to-Lock position and isolating the discharge flow path of the pump from the RCS by a least one closed isolation valve. Use of the alternative method requires inoperability verification ~~at least once every 12 hours~~ INSERT 1.

REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

## LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of one vent valve and one block valve powered from emergency busses shall be OPERABLE and closed*at each of the following locations:

a. Reactor vessel head, and

b. Pressurizer steam space.

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

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves and block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

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

## SURVEILLANCE REQUIREMENTS

4.4.11.1 Each Reactor Coolant System vent path block valve not required to be closed by ACTION a. or b., above, shall be demonstrated OPERABLE at least once per COLD SHUTDOWN, if not performed within the previous 92 days, by operating the valve through one complete cycle of full travel from the control room.

4.4.11.2 Each Reactor Coolant System vent path shall be demonstrated OPERABLE ~~at least once per 18 months~~ by:

INSERT 1

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

*For an OPERABLE vent path using a power-operated relief valve (PORV) as the vent path, the PORV block valve is not required to be closed.

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SURVEILLANCE REQUIREMENTS

4.4.11.2 (Continued)

b. Cycling each vent valve through at least one complete cycle of full travel from the control room, and

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

## 3/4.5 EMERGENCY CORE COOLING SYSTEMS

### 3/4.5.1 ACCUMULATORS

### HOT STANDBY, STARTUP, AND POWER OPERATION

### LIMITING CONDITION FOR OPERATION

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

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6121 and 6596 gallons,

c. A boron concentration of between the limits specified in the COLR, and

d. A nitrogen cover-pressure of between 585 and 664 psig.

APPLICABILITY: MODES 1, 2, and 3*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

c. With one pressure or water level channel inoperable per accumulator, return the inoperable channel to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With two pressure channels or two water level channels inoperable per accumulator, immediately declare the affected accumulator(s) inoperable.

### SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. ~~At least once per 24 hours~~ by: INSERT 1

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS

4.5.1.1 (Continued)

2) Verifying that each accumulator isolation valve is open.

b. By verifying the boron concentration of the accumulator solution under the following conditions:

1) At ~~least once~~ ~~per 31~~ days

INSERT 1

2) Within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume. This surveillance is not required when the volume increase makeup source is the RWST and the RWST has not been diluted since verifying that the RWST boron concentration is equal to or greater than the accumulator boron concentration limit.

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

d. ~~At least once per~~ 18 ~~months~~ by verifying that each accumulator isolation valve opens automatically under each of the following conditions:

1) When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and

2) Upon receipt of a Safety Injection test signal.

EMERGENCY CORE COOLING SYSTEMS

ACCUMULATORS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each reactor coolant system accumulator isolation valve shall be shut with power removed from the valve operator.

APPLICABILITY: MODES 4* and 5**.

ACTION:

With one or more accumulator isolation valve(s) open and/or power available to the valve operator(s), immediately close the accumulator isolation valves and/or remove power from the valve operator(s).

SURVEILLANCE REQUIREMENTS

4.5.1.2 Each accumulator isolation valve will be verified shut with power removed from the valve operator ~~at least once per 31 days~~.

INSERT 1

*Within 12 hours prior to entry into MODE 3 from MODE 4 and if pressurizer pressure is greater than 1000 psig, each accumulator isolation valve shall be open as required by Specification 3.5.1.1.a.

**With accumulator pressure greater than 100 psig.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

INSERT 1

a. ~~At least once per 24 hours~~ by verifying that the following valves are in the indicated positions with power to the valve operators removed:

| Valve Number | Valve Function | Valve Position |
|---|---|---|
| SI-V-3 | Accumulator Isolation | Open* |
| SI-V-17 | Accumulator Isolation | Open* |
| SI-V-32 | Accumulator Isolation | Open* |
| SI-V-47 | Accumulator Isolation | Open* |
| SI-V-114 | SI Pump to Cold-Leg Isolation | Open |
| RH-V-14 | RHR Pump to Cold-Leg Isolation | Open |
| RH-V-26 | RHR Pump to Cold-Leg Isolation | Open |
| RH-V-32 | RHR to Hot-Leg Isolation | Closed |
| RH-V-70 | RHR to Hot-Leg Isolation | Closed |
| SI-V-77 | SI to Hot-Leg Isolation | Closed |
| SI-V-102 | SI to Hot-Leg Isolation | Closed |

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

INSERT 1

1) Verifying that the ECCS piping is full of water, and ⊗

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment 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 primary CONTAINMENT INTEGRITY, and

2) At least once daily of the areas affected within containment by containment entry and during the final entry when primary CONTAINMENT INTEGRITY is established.

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS

4.5.2 (Continued)

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

INSERT 1

1) Verifying automatic interlock action of the RHR system from the Reactor Coolant System to ensure that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 440 psig, the interlocks prevent the valves from being opened.

2) 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 abnormal 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 (Safety Injection actuation and Automatic Switchover to Containment Sump) test signals, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying OPERABILITY of each pump when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump;

2) Safety Injection pump; and

3) RHR pump.

4.5.2 (Continued)

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

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

INSERT 1

| High Head SI System Valve Number | Intermediate Head SI System Valve Number |
|---|---|
| SI-V-143 | SI-V-80 |
| SI-V-147 | SI-V-85 |
| SI-V-151 | SI-V-104 |
| SI-V-155 | SI-V-109 |
| | SI-V-117 |
| | SI-V-121 |
| | SI-V-125 |
| | SI-V-129 |

h. NOT USED

(4)

## SURVEILLANCE REQUIREMENTS

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

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

INSERT 1

*An additional charging pump may be made capable of injecting under administrative control for up to 1 hour during pump-swap operation, except during RCS water-solid conditions. Additionally, an inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

**An alternate method to assure pump inoperability may be used by placing the control room pump-control switch(s) in the Pull-to-Lock position and isolating the discharge flow path of the pump(s) from the RCS by at least one closed isolation valve. Use of the alternate method requires inoperability verification ~~at least once every 12 hours~~.

INSERT 1

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{avg}$ EQUAL TO OR LESS THAN 200°F

LIMITING CONDITION FOR OPERATION

3.5.3.2 As a minimum, the following number of Safety Injection pumps shall be inoperable*:

a. Two when the RCS vent area is less than 18 square inches.

b. One when the RCS vent area is equal to or greater than 18 square inches,
or

c. One when the RCS is in a reduced inventory condition**.

APPLICABILITY: MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned.

ACTION:

With fewer than the required number of Safety Injection pumps inoperable, immediately restore all pumps required to inoperable status.

SURVEILLANCE REQUIREMENTS

4.5.3.2 All Safety Injection pumps required to be inoperable shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position ~~at least once per 31 days~~***.

INSERT 1

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

** A reduced inventory condition exists whenever reactor vessel (RV) water level is lower than 36 inches below the RV flange.

*** An alternate method to assure pump inoperability may be used by placing the control room pump-control switch(s) in the Pull-to-Lock position and isolating the discharge flow path of the pump(s) from the RCS by at least one closed isolation valve. Use of the alternate method requires inoperability verification ~~at least once every 12 hours~~.



INSERT 1

BORON INJECTION SYSTEM

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A minimum contained borated water volume of 477,000 gallons,

b. A boron concentration between the limits specified in the COLR,

c. A minimum solution temperature of 50°F, and

d. A maximum solution temperature of 98°F.

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

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. ~~At least once per 7 days~~ by:

1) Verifying the contained borated water volume in the tank, and

INSERT 1

2) Verifying the boron concentration of the water.

b. ~~At least once per 24 hours~~ by verifying the RWST temperature.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

INSERT 1

a. ~~At least once per 31 days~~ by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions except for valves that are open under administrative control as permitted by Specification 3.6.3; and

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

SURVEILLANCE REQUIREMENTS

---

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. With the leakage rate in accordance with the Containment Leakage Rate Testing Program.

b. At ~~least~~ once ~~per~~ 24 ~~months~~ by verifying that only one door in each air lock can be opened at a time.

INSERT 1

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between 14.6 and 16.2 psia.

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

PRIMARY CONTAINMENT

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined ~~at least once per 24 hours~~:

INSERT 1

Location

a. Elevation 45 feet

b. Elevation 71 feet

c. Elevation 110 feet

d. Elevation 130 feet

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

CONTAINMENT VENTILATION SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is in accordance with the Containment Leakage Rate Testing Program.

4.6.1.7.2 Each 8-inch containment purge supply and exhaust isolation valve shall be verified to be sealed closed or open in accordance with Specification 3.6.1.7 ~~at least once per 31 days~~

INSERT 1

CONTAINMENT SYSTEMS

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 automatically transferring suction to the containment sump.

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

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

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

INSERT 1

b. By verifying OPERABILITY of each pump 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 Pressure-Hi-3 test signal, and

2) Verifying that each spray pump starts automatically on a Containment Pressure-Hi-3 test signal.

d. By verifying each spray nozzle is unobstructed following activities that could result in nozzle blockage.

*In MODE 4, when the Residual Heat Removal System is in operation, an OPERABLE flow path is one that is capable of taking suction from the refueling water storage tank upon being manually realigned.

CONTAINMENT SYSTEMS

DEPRESSURIZATION AND COOLING SYSTEMS

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 9420 and 9650 gallons of between 19 and 21% by weight NaOH solution, and

b. Two gravity feed paths each capable of adding NaOH solution from the chemical additive tank to the Refueling Water Storage Tank.

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 contained solution volume in the tank, and

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

c. ~~At least once per 18 months~~, during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal.

## 3/4.6.3 CONTAINMENT ISOLATION VALVES

### LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE*.

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

ACTION:

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

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

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

### SURVEILLANCE REQUIREMENTS

4.6.3.1 Not used

4.6.3.2 Each containment isolation valve shall be demonstrated OPERABLE during shutdown ~~at least once per 18 months~~ by:

INSERT 1

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

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

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

CONTAINMENT SYSTEMS

COMBUSTIBLE GAS CONTROL

HYDROGEN MIXING SYSTEM

## LIMITING CONDITION FOR OPERATION

3.6.4.3 Two independent Containment Structure Recirculation Fan Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Containment Structure Recirculation Fan inoperable, restore the inoperable fan to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

## SURVEILLANCE REQUIREMENTS

4.6.4.3 Each Containment Structure Recirculation Fan System shall be demonstrated OPERABLE:

INSERT 1

a. ~~At least once per 92 days on a STAGGERED TEST BASIS~~ by starting each system from the control room and verifying that the system operates for at least 15 minutes, and

b. ~~At least once per 18 months~~ by verifying a system flow rate of at least 4000 cfm through the hydrogen mixing flow path.

## 3/4.6.5 CONTAINMENT ENCLOSURE BUILDING

### CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANUP SYSTEM

### LIMITING CONDITION FOR OPERATION

3.6.5.1 Two independent Containment Enclosure Emergency Air Cleanup Systems shall be OPERABLE.

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

ACTION:

With one Containment Enclosure Emergency Air Cleanup System inoperable, restore the inoperable system to OPERABLE status within 7 days# or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.5.1 Each Containment Enclosure Emergency Air Cleanup System shall be demonstrated OPERABLE:

INSERT 1

a. ~~At least once per 31 days on a STAGGERED TEST BASIS~~ by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

b. ~~At least once per 18 months~~ or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978*, and the system flow rate is 2100 cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or

# The 7-day allowed outage time which was entered on June 4, 2006 at 0602 hours, may be extended one time by an additional 7 days to complete repair and testing on the Containment Enclosure Ventilation Area return fan EAH-FN-31B.

* ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Rev. 2, March 1978.

CONTAINMENT SYSTEMS

CONTAINMENT ENCLOSURE BUILDING

CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANUP SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.5.1b.2 (Continued)

equal to 5% when tested at a temperature of 30°C, at a relative humidity of 95% and a face velocity of 46 fpm in accordance with ASTM-D3803-1989; and

3) Verifying a system flow rate of 2100 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 5% when tested at a temperature of 30°C, at a relative humidity of 95% and a face velocity of 46 fpm in accordance with ASTM-D3803-1989.

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

INSERT 1

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 2100 cfm ± 10%,

2) Verifying that the system starts on a Safety Injection test signal,

3) Verifying that the filter cross connect valves can be manually opened, and

4) Verifying that each system produces a negative pressure of greater than or equal to 0.25 inch Water Gauge in the annulus within 4 minutes after a start signal.

e. After each complete or partial replacement of a high efficiency particulate air (HEPA) filter bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a dioctyl phthalate (DOP) test aerosol while operating the system at a flow rate of 2100 cfm ± 10%; and

CONTAINMENT SYSTEMS

CONTAINMENT ENCLOSURE BUILDING

CONTAINMENT ENCLOSURE BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.2 Containment enclosure building integrity shall be maintained.

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

--------------------NOTE--------------------

Entry into ACTION is not required when the access opening is being used for normal transit entry or exit.

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

ACTION:

a. Without containment enclosure building integrity for reasons other than Action b, restore containment enclosure building integrity within 12 hours. Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Without containment enclosure building integrity when equipment ingress and egress requires the access door to be maintained open, verify a dedicated individual, who is in continuous communication with the control room, is available to rapidly close the door; and restore containment enclosure building integrity within 24 hours. Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Containment enclosure building integrity shall be demonstrated:

INSERT 1

a. ~~At least once per 31 days by~~ verifying that the door in each access opening is closed except when the access opening is being used for normal transit entry and exit, and

b. ~~At least once per 36 months on a STAGGERED TEST BASIS~~ by verifying the containment enclosure building can be maintained at a negative pressure greater than or equal to 0.25 inch water gauge by one train of the containment enclosure emergency air cleanup system within 4 minutes after a start signal.

PLANT SYSTEMS

TURBINE CYCLE

AUXILIARY FEEDWATER SYSTEM

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

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

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) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and

3) Verifying that valves FW-156 and FW-163 are OPERABLE for alignment of the startup feedwater pump to the emergency feedwater header.

INSERT 1

b. At ~~least~~ once ~~per~~ 92 days ~~on a STAGGERED TEST BASIS~~ by verifying the following pumps develop the required discharge pressure and flow as specified in the Technical Requirements Manual:

1) The motor-driven emergency feedwater pump;

2) The steam turbine-driven emergency feedwater pump when the secondary steam supply pressure is greater than 500 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

3) The startup feedwater pump.

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 upon receipt of an Emergency Feedwater System Actuation test signal;

2) Verifying that each emergency feedwater pump starts as designed automatically upon receipt of an Emergency Feedwater Actuation System test signal;

No changes - provided for information only



4.7.1.2.1 c. (Continued)

3) Verifying that with all manual actions, including power source and valve alignment, the startup feedwater pump starts within the required elapsed time; and

4) Verifying that each emergency feedwater control valve closes on receipt of a high flow test signal.

4.7.1.2.2 Auxiliary feedwater flow paths to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days, or after maintenance on an auxiliary feedwater pump that could have an effect upon pump performance, prior to entering MODE 2 by verifying normal flow to each steam generator from:

a. Each emergency feedwater pump, and

b. The startup feedwater pump via the main feedwater flow path and via the emergency feedwater header.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) system shall be OPERABLE with

a. A volume of 212,000 gallons of water contained in the condensate storage tank, and

b. A concrete CST enclosure that is capable of retaining 212,000 gallons of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the CST or the CST enclosure inoperable, within 4 hours restore the CST and the CST enclosure to OPERABLE status 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

4.7.1.3a. The CST shall be demonstrated OPERABLE ~~at least once per 12 hours~~ by verifying the contained water volume in the CST is within its limits.

b. The CST enclosure shall be demonstrated OPERABLE ~~at least once per 18 months~~ by an inspection to verify that CST enclosure integrity is maintained.

PLANT SYSTEMS

TURBINE CYCLE

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant shall be less than or equal to 0.1 μCi/gm DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the secondary coolant greater than 0.1 μCi/gm DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.7.1.4 At ~~least once every 31~~ days verify the specific activity of the secondary coolant is less than or equal to 0.1 μCi/gm DOSE EQUIVALENT I-131.

*The provisions of Specification 4.0.4 are not applicable for entry into MODE 4, however, once steam generator pressure exceeds 100 psig, the requirements of Specification 4.7.1.4 must be met within 12 hours if not performed within the past 31 days.

ATMOSPHERIC RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 At least four atmospheric relief valves and associated manual controls including the safety-related gas supply systems shall be OPERABLE.

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

ACTION:

a. With one less than the required atmospheric relief valves OPERABLE, restore the required atmospheric relief valves to OPERABLE status within 7 days; or be in at least HOT STANDBY within the next 12 hours.

b. With two less than the required atmospheric relief valves OPERABLE, restore at least three atmospheric relief valves to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric relief valve and associated manual controls including the safety-related gas supply systems shall be demonstrated OPERABLE:

a. INSERT 1 ~~At least once per 24 hours~~ by verifying that the nitrogen accumulator tank is at a pressure greater than or equal to 500 psig.

b. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, verify that all valves will open and close fully by operation of manual controls.

*When steam generators are being used for decay heat removal.

#Entry into this MODE is permitted for up to 24 hours to perform post-modification or post-maintenance testing to verify OPERABILITY of components. ACTION requirements shall not apply until OPERABILITY has been verified.

## 3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent primary component cooling water loops shall be OPERABLE, including one OPERABLE pump in each loop.

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

ACTION:

With one primary component cooling water (PCCW) loop inoperable, restore the required primary component cooling water loop 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 primary 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; and

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 its associated Engineered Safety Feature actuation signal.

## 3/4.7.4 SERVICE WATER SYSTEM/ULTIMATE HEAT SINK

### SURVEILLANCE REQUIREMENTS

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

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

b. At ~~least once per~~ 18 ~~months~~ during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal.

INSERT 1

4.7.4.2 Each service water cooling tower loop shall be demonstrated OPERABLE:

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

b. At ~~least once per~~ 18 ~~months~~ during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal,

2) Each automatic valve in the flowpath actuates to its correct position on a Tower Actuation (TA) test signal and

3) Each service water cooling tower pump starts automatically on a TA signal.

4.7.4.3 The service water pumphouse shall be demonstrated OPERABLE ~~at least once per 24 hours~~ by verifying the water level to be at or above 25.1' (-15.9' Mean Sea Level). H

4.7.4.4 The mechanical draft cooling tower shall be demonstrated OPERABLE:

INSERT 1

a. At ~~least once per 24 hours~~ by verifying the water in the mechanical draft cooling tower basin to be at a level of greater than or equal to 42.15* feet.

b. ~~At least once per week~~ by verifying that the water in the cooling tower basin to be at a bulk average temperature of less than or equal to 70°F.

*With the cooling tower in operation with valves aligned for tunnel heat treatment, the tower basin level shall be maintained at greater than or equal to 40.55 feet.

SURVEILLANCE REQUIREMENTS

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

INSERT 1

1) Starting from the control room each cooling tower fan that is required to be OPERABLE and operating each of these fans for at least 15 minutes, and

2) Verifying that the portable tower makeup pump system is stored in its design operational readiness state.

d. ~~At least once per 18 months~~ by verifying that the portable tower makeup pump develops a flow greater than or equal to 200 gpm.

## 3/4.7.6 CONTROL ROOM SUBSYSTEM

## EMERGENCY MAKEUP AIR AND FILTRATION

### LIMITING CONDITION FOR OPERATION (Continued)

In MODE 5 or 6, or during movement of irradiated fuel assemblies:

d. With one CREMAFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable system to OPERABLE status within 7 days or either immediately initiate and maintain operation of the remaining OPERABLE CREMAFS train in the filtration/recirculation mode or immediately suspend movement of irradiated fuel assemblies.

e. With both CREMAFS trains inoperable, or with the OPERABLE CREMAFS train, required to be in the filtration/recirculation mode by ACTION d., not capable of being powered by an OPERABLE emergency power source, immediately suspend all movement of irradiated fuel assemblies.

f. With one or both CREMAFS trains inoperable due to an inoperable CRE boundary, immediately suspend movement of irradiated fuel assemblies.

### SURVEILLANCE REQUIREMENTS

4.7.6.1 Each CREMAFS train shall be demonstrated OPERABLE:

a. At ~~least~~ once ~~per 31~~ days ~~on a STAGGERED~~ TEST ~~BASIS~~ by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

INSERT 1

PLANT SYSTEMS

CONTROL ROOM SUBSYSTEMS

EMERGENCY MAKEUP AIR AND FILTRATION

SURVEILLANCE REQUIREMENTS (Continued)

b. ~~At least once per 18 months~~ or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

INSERT 1

1) Verifying that the filtration system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than .05% and uses the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978*, and the system flow rate is 1100 cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 34.5 fpm (Train A) and 58.3 fpm (Train B) in accordance with ASTM-D-3803-1989;

3) Verifying a system flow rate of 1100 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 34.5 fpm (Train A) and 58.3 fpm (Train B) in accordance with ASTM-D-3803-1989;

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

INSERT 1

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks, for filter CBA-F-38, is less than 2.8 inches Water Gauge while operating the system at a flow rate of 1100 cfm ± 10%; and verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks, for filter CBA-F-8038, is less than 6.3 inches Water Gauge while operating the system at a flow rate of 1100 cfm ± 10%.

*ANSI N510-1980 shall be used in place of ANSI N510-1975 as referenced in Regulatory Guide 1.52, Revision 2, March 1978.

no changes - provided for information only



SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying that upon generation of an 'S' test signal, the following automatic system functions occur:

   a. The normal makeup air fan(s) trip off and the normal makeup air isolation damper(s) close;

   b. The control room exhaust subsystem isolation damper(s) close, and the exhaust fan trips off;

   c. The control room emergency makeup air and filtration subsystem actuates with flows through the HEPA filters and charcoal adsorber banks;

3) Verifying that upon generation of Remote Intake High Radiation test signal, the following automatic system functions occur:

   a. The normal makeup air fan(s) trip off and the normal makeup air isolation damper(s) close;

   b. The control room exhaust subsystem isolation damper(s) close, and the exhaust fan trips off;

   c. The control room emergency makeup air and filtration subsystem actuates with flows through the HEPA filters and charcoal adsorber banks;

4) Verifying that the heaters dissipate at least 3.24 kW (based on design rated voltage of 460V) when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the filtration system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than .05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 1100 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the filtration system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than .05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 1100 cfm ± 10%.

g. By performing required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

PLANT SYSTEMS

3/4.7.6 CONTROL ROOM SUBSYSTEMS

AIR CONDITIONING

LIMITING CONDITION FOR OPERATION

3.7.6.2 Two independent Control Room Air Conditioning Subsystems shall be OPERABLE.

APPLICABILITY: All MODES

ACTION:

MODES 1, 2, 3 and 4:

With one Control Room Air Conditioning Subsystem inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one Control Room Air Conditioning Subsystem inoperable, restore the inoperable system to OPERABLE status within 30 days or initiate and maintain operation of the remaining OPERABLE Control Room Air Conditioning Subsystem or immediately suspend all operations involving CORE ALTERATION.

b. With both Control Room Air Conditioning Subsystems inoperable, or with the OPERABLE Control Room Air Conditioning Subsystem unable to maintain temperature below the limiting equipment qualification temperature in the control room area, suspend all operations involving CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.7.6.2 Each Control Room Air Conditioning Subsystem shall be demonstrated OPERABLE ~~at least once per 92 days~~ by verifying the ability to maintain temperature in the control room area below the limiting equipment qualification temperature for 24 hours.

INSERT 1

## 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 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of 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 microCurie 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.

INSERT 1

a. Sources in use - At ~~least once~~ per ~~6 months~~ for all sealed sources containing radioactive materials:

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

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

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

b. Demonstrated OPERABLE at ~~least once per~~ 18 ~~months~~ by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.*

INSERT 1

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE.**

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

1) Verifying the fuel level in the day fuel tank;

2) Verifying the fuel level in the fuel storage tank;

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank;

4) Verifying the lubricating oil inventory in storage;

5) Verifying the diesel starts from standby conditions and attains a steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.***

* This surveillance requirement shall not be performed in Mode 1 or 2.

** All planned starts for the purpose of these surveillances may be preceded by an engine prelube period.

*** A modified start involving idling and gradual acceleration to synchronous speed may be used for this surveillance. When modified start procedures are not used, the time, voltage, and frequency tolerances of Specification 4.8.1.1.2e must be met.

4.8.1.1.2 (Continued)

6) Verifying the generator is synchronized, gradually loaded**** to greater than or equal to 5600 kW and less than or equal to 6100 kW, and operates within this load band for at least 60 minutes, and until stable engine operating temperature is attained; and

7) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At ~~least once per 31 days~~ by checking for and removing accumulated water from the day fuel tank;

INSERT 1

c. At ~~least once per 31 days~~ by checking for and removing accumulated water from the fuel oil storage tanks;

d. By verifying fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program;

e. At least ~~once every 184~~ days# by verifying the diesel starts from standby condition and achieves:

1) A generator voltage and frequency greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the start signal, and

2) A steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.

**** Diesel generator loading may be in accordance with manufacturers recommendations, including a warmup period. Momentary transients outside the load range, due to changing bus conditions, do not invalidate the test. In addition, this surveillance shall be preceded by and immediately follow without shutdown a successful performance of Specification 4.8.1.1.2a.5) or 4.8.1.1.2e.

# Performance of Specification 4.8.1.1.2a.6) must immediately follow this surveillance. Additionally, performance of Specification 4.8.1.1.2e satisfies Specification 4.8.1.1.2a.5).

4.8.1.1.2 (Continued)

f. ~~At least once per~~ 18 ~~months~~, during shutdown##, by:

INSERT 1

1) (NOT USED)

2) Verifying the generator capability to reject a load of greater than or equal to 671 kW while maintaining voltage at 4160 $\pm$ 420 volts and frequency at 60 $\pm$ 4.0 Hz;

3) Verifying the generator capability to reject a load of 6083 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection;

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b) Verifying the diesel starts from standby conditions### on the loss of offsite power signal, energizes the emergency busses with permanently connected loads within 12 seconds, energizes the auto-connected shutdown loads through the emergency power sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on an SI actuation test signal, without loss-of-offsite power, the diesel generator starts from standby conditions### on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test;

## Selected surveillance requirements, or portions thereof, may be performed during conditions or modes other than shutdown, provided an evaluation supports safe conduct of that surveillance in a condition or mode that is consistent with safe operation of the plant. (Ref. NRC GL 91-04)

### Starting of the diesel for Specifications 4.8.1.1.2f.4) and 4.8.1.1.2f.5) may be performed with the engine at or near normal operating temperature.

No changes - provided for information only

4.8.1.1.2 (Continued)

6) Simulating a loss-of-offsite power in conjunction with an SI actuation test signal; and

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

   b) Verifying the diesel starts from standby conditions, on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the emergency power sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test; and

   c) Verifying that all automatic diesel generator trips, except engine overspeed, low lube oil pressure, 4160-volt bus fault, and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection actuation signal.

7) Verifying full-load carrying capability of the diesel generator for an interval of not less than 24 hours:

   a) At a load greater than or equal to 5600 kW and less than or equal to 6100 kW,[####] or

   b) Should auto-connected loads be equal to or greater than 6100 kW;

      1. Verify the diesel generator operates for an interval of not less than 2 hours at a load greater than or equal to 6363 kW and less than or equal to 6700 kW.[####] For the remaining hours, at a load greater than or equal to 5600 kW and less than or equal to 6100 kW, and

      2. Verify that the auto-connected loads to each diesel generator do not exceed the short time rating of 6700 kW.

#### Diesel generator loading may be in accordance with manufacturers recommendations, including a warmup period. Momentary transients outside the load range, due to changing bus conditions, do not invalidate the test.

No changes - provided for information only

SURVEILLANCE REQUIREMENTS

4.8.1.1.2 (Continued)

8) Within 5 minutes of shutting down the diesel generator, after the diesel generator has operated for an interval of not less than 2 hours at a load greater than or equal to 5600 kW and less than or equal to 6100 kW,+ by verifying the diesel starts and achieves:

   a) A generator voltage and frequency greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the start signal, and

   b) A steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.

9) Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines;

12) Verifying that the emergency power sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval;

13) NOT USED

+ Momentary transients outside the load range, due to changing bus conditions, do not invalidate the test.

4.8.1.1.2 (Continued)

14) Simulating a Tower Actuation (TA) signal while the diesel generator is loaded with the permanently connected loads and auto-connected emergency (accident) loads, and verifying that the service water pump automatically trips, and that the cooling tower pump automatically starts. After energization the steady state voltage and frequency of the emergency buses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz; and

15) While diesel generator 1A is loaded with the permanently connected loads and auto-connected emergency (accident) loads, manually connect the 1500 hp startup feedwater pump to 4160-volt bus E5. After energization the steady-state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz.

g. At ~~least once per~~ 10 ~~years~~ or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously from standby condition, during shutdown, and verifying that both diesel generators achieve:

INSERT 1

1) A generator voltage and frequency greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the start signal, and

2) A steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

---

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE and energized:

a. Train A

1) 125-volt Battery Banks 1A and 1C,
2) One full-capacity battery charger on Bus #11A, and
3) One full-capacity battery charger on Bus #11C.

b. Train B

1) 125-volt Battery Banks 1B and 1D,
2) One full-capacity battery charger on Bus #11B, and
3) One full-capacity battery charger on Bus #11D.

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

ACTION:

a. With one of the required battery banks in one train inoperable, close the bus tie to connect the remaining operable battery bank to the D.C. bus supplied by the inoperable battery bank within 2 hours; restore the inoperable battery bank to OPERABLE status within 30 days* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the full-capacity chargers inoperable, restore the inoperable charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

---

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

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

1) The parameters in Table 4.8-2 meet the Category A limits, and
2) The total battery terminal voltage is greater than or equal to 128 volts on float charge.

INSERT 1

b. At ~~least once~~ per ~~92 days~~ and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

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*No more than one battery at a time may be taken out of service for more than 30 days.

4.8.2.1b (Continued)

1) The parameters in Table 4.8-2 meet the Category B limits,

2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150 x $10^{-6}$ ohm,* and

3) The average electrolyte temperature of 16 connected cells (4 cells per row) is above 65°F.

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

INSERT 1

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

2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150 x $10^{-6}$ ohm,* and

4) Each battery charger will supply at least 150 amperes at a minimum of 132 volts for at least 8 hours.

d. At ~~least once per 18 months~~ by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test;

e. At least ~~once~~ per 60 ~~months~~ by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

* Obtained by subtracting the normal resistance of: (1) the cross room rack connector (210 x $10^{-6}$ ohm, typical) and (2) the bi-level rack connector (35 x $10^{-6}$ ohm, typical) from the measured cell-to-cell conection resistance.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses and panels shall be determined energized in the required manner ~~at least once per 7 days~~ by verifying correct breaker alignment and indicated voltage on the bus~~s~~es.

INSERT 1

---

*No more than one Battery Bank (1A, 1B, 1C, or 1D) at a time may be taken out of service for more than 30 days.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

## LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical bus~~s~~es shall be energized in the specified manner:

a. One train of A.C. emergency bus~~s~~es consisting of the 4160-volt and the 480-volt A.C. emergency bus~~s~~es listed in 3.8.3.1a. and b. (excluding 480-volt Emergency Bus #E64);

b. Two of the four 120-volt A.C. vital Panels 1A, 1B, 1C, and 1D energized from their associated inverters connected to their respective D.C. bus~~s~~es;

c. One of the two 120-volt A.C. Vital Panels 1E or 1F energized from its associated inverter connected to the respective D.C. bus; and

d. Two 125-volt D.C. bus~~s~~es (in the same train) energized from their associated battery banks.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical bus~~s~~es and panels not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical bus~~s~~es and panels in the specified manner as soon as possible, and within 8 hours, depressurize and vent the RCS through at least a 1.58-square-inch vent.

## SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified bus~~s~~es and panels shall be determined energized in the required manner at ~~least once per~~ 7 days by verifying correct breaker alignment and indicated voltage on the busses.

INSERT 1

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

TRIP CIRCUIT FOR INVERTER I-2A

## LIMITING CONDITION FOR OPERATION

3.8.3.3 The safety-related trip circuit that trips the D.C. feed from D.C. Bus #11C to inverter #I-2A after 15 minutes of discharge from the battery shall be OPERABLE. Note that this LIMITING CONDITION FOR OPERATION is applicable only when D.C. Bus #11C is required to be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6.

ACTION:

With this safety-related trip circuit inoperable, restore the trip circuit to OPERABLE status within 7 days or de-energize the D.C. feed to inverter #I-2A by tripping the D.C. circuit breaker in D.C. Bus #11C. Verify that this breaker is open once per 7 days thereafter.

## SURVEILLANCE REQUIREMENTS

4.8.3.3 The safety-related trip circuit shall be demonstrated operable at least once per 18 months.

INSERT 1

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

## LIMITING CONDITION FOR OPERATION

3.8.4.1 The circuit breakers feeding the following loads inside primary containment shall be padlocked in the open position:

| Loads | Circuit | Panel |
|---|---|---|
| Refueling Canal Skimmer Pump | 1-SF-P-272 | 1-ED-MCC-111 |
| Polar Gantry Crane | 1-MM-CR-3 | 1-ED-US-11 |
| Distribution Panel | 1-ED-PP-7A | 1-ED-US-11 |
| Distribution Panel | 1-ED-PP-7B | 1-ED-US-23 |
| Rod Control Cluster Change Fixture | 1-FH-RE 12 | 1-ED-MCC-111 |

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

EXCEPTION:

If any of the above-mentioned loads are required for brief durations (not to exceed 72 hours) during plant operation, the pertinent circuit breaker can be unlocked and closed for the required duration provided this change in breaker position becomes part of the applicable operating procedure used for the work inside containment.

## SURVEILLANCE REQUIREMENTS

4.8.4.1 Verify ~~at least once per 31 days~~ that the circuit breakers listed above are locked in the open position.

INSERT 1

ELECTRICAL POWER SYSTEMS

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

## LIMITING CONDITION FOR OPERATION

3.8.4.3 Each thermal overload protection for safety-related motor-operated valves shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above-required valves inoperable, bypass the inoperable thermal overload within 8 hours, restore the inoperable thermal overload to OPERABLE status within 30 days, or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected system(s).

## SURVEILLANCE REQUIREMENTS

INSERT 1

4.8.4.3 The thermal overload protection for the above required valves shall be demonstrated OPERABLE ~~at least once per 18 months~~ and following maintenance on the motor starter by selection of a representative sample of at least 25% of all thermal overloads for the above-required valves and replacing them with precalibrated devices that have been subjected to a CHANNEL CALIBRATION.

## 3/4.9 REFUELING OPERATIONS

## 3/4.9.1 BORON CONCENTRATION

### LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure a boron concentration of greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration equivalent to 30 gpm at a boron concentration greater than or equal to the limit specified in the COLR for the Boric Acid Storage System until the boron concentration is restored to greater than or equal to the limit specified in the COLR.

### SURVEILLANCE REQUIREMENTS

4.9.1.1 Verify boron concentration is within the limits specified in the COLR prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at le~~ast once~~ per ~~72 hours~~

INSERT 1

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

## 3/4.9.2 INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, immediately initiate corrective action to restore one source range neutron flux monitor to OPERABLE status and determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

### SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at ~~least once~~ per ~~12 hours~~, INSERT 1

b. A CHANNEL CALIBRATION* at least ~~once~~ per ~~18 months~~.

*Neutron detectors may be excluded from CHANNEL CALIBRATION.

## SURVEILLANCE REQUIREMENTS

4.9.4 For the above required containment building penetrations:

a. Determine that each of the above required containment building penetrations shall be in its required condition within 100 hours prior to the start of, and ~~at least once per 7 days~~ during movement of recently irradiated fuel in the containment building, ~~and~~ INSERT 1

b. Demonstrate that the Containment Purge and Exhaust Isolation System is OPERABLE ~~at least once every 18 months~~ and within 10 days prior to the start of movement of recently irradiated fuel in the containment building by verifying that containment purge and exhaust isolation occurs on manual initiation and on a High Radiation test signal from each of the manipulator crane radiation area monitoring instrumentation channels. **

** Not required for those valves complying with Specification 3.9.4.c.1 or Specification 3.9.4.c.3.

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2750 gpm ~~at least once per 12 hours~~.

INSERT 1

* The RHR 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 vessel hot legs.

REFUELING OPERATIONS

RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR 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. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2750 gpm at ~~least once~~ per ~~12 hours~~.

INSERT 1

* Prior to initial criticality, the RHR 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 vessel hot legs.

REFUELING OPERATIONS

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 vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

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

INSERT 1

## REFUELING OPERATIONS

## 3/4.9.11 WATER LEVEL - STORAGE POOL

### LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

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

b. The provisions of Specification 3.0.3 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 STORAGE BUILDING EMERGENCY AIR CLEANING SYSTEM

### LIMITING CONDITION FOR OPERATION

3.9.12 Two independent trains of the Fuel Storage Building Emergency Air Cleaning System shall be OPERABLE whenever irradiated fuel is in the storage pool and shall be OPERABLE with one train operating during fuel movement.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one train of the Fuel Storage Building Emergency Air Cleaning System inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE train of the Fuel Storage Building Emergency Air System is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no trains of the Fuel Storage Building Emergency Air Cleaning System OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one train of the Fuel Storage Building Emergency Air Cleaning System is restored to OPERABLE status and is in operation.

c. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.12 The above required trains of the Fuel Storage Building Emergency Air Cleaning System shall be demonstrated OPERABLE:

INSERT 1

a. At least ~~once~~ per 31 da~~ys on a~~ STAGGE~~RED TEST~~ ~~BASIS~~ by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least ~~once~~ per 18 ~~months~~ or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

4.9.12b (Continued)

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,* and the system flow rate is 16,450 cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 44 fpm in accordance with ASTM-D-3803-1989; and

3) Verifying a system flow rate of 16,450 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than or equal to 2.5% when tested at a temperature of 30°C, at a relative humidity of 70% and a face velocity of 44 fpm in accordance with ASTM-D-3803-1989.

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

INSERT 1

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 16,450 cfm ± 10%,

2) Verifying that the system maintains the spent fuel storage pool area at a negative-pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere during system operation,

* ANSI N510-1980 shall be used in place of ANSI N510-1975 as referenced in Regulatory Guide 1.52, Rev. 2, March 1978.

No changes
Provided for information only



4.9.12d (Continued)

3) Verifying that the filter cross connect valve can be manually opened, and

4) Verifying that the heaters dissipate at least 84 kW (based on design rated voltage of 480V) when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 16,450 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 16,450 cfm ± 10%.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

## LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

## SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined ~~at least once per 2 hours~~. INSERT 1

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

## 3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

### LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

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

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

### SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER ~~at least once per hour~~ during PHYSICS TESTS.

INSERT 1

4.10.2.2 The requirements of Specifications 4.2.2.2, 4.2.2.3, and 4.2.3.2 shall be performed ~~at least once per 12 hours~~ during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

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

b. The Reactor Trip Setpoints on the OPERABLE Intermediate Range and Power Range* 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 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

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

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER ~~at least once per hour~~ during PHYSICS TESTS.

4.10.3.2 Verify each OPERABLE Intermediate Range and Power Range* channel has been subjected to an ANALOG CHANNEL OPERATIONAL TEST per Specification Table 4.3-1 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 541°F ~~at least once per 30 minutes~~ during PHYSICS TESTS.

INSERT 1

*Power Range Low Setpoint only.

SPECIAL TEST EXCEPTIONS

## 3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

### SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined ~~to~~ be INSERT 1 OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps when the rods are stationary.

*This requirement is not applicable during the initial calibration of the Digital Rod Position Indication System provided: (1) $k_{eff}$ is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

RADIOACTIVE EFFLUENTS

LIQUID EFFLUENTS

LIQUID HOLDUP TANKS*

## LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each temporary unprotected outdoor 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 temporary unprotected outdoor tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.8.1.4.

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 temporary unprotected outdoor 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 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.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREMAFS, operating at a flow rate of less than or equal to 600 CFM at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air in-leakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air in-leakage measured by the testing described in paragraph c. The unfiltered air in-leakage limit for radiological challenges is the in-leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered in-leakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

m. Reactor Coolant Pump Flywheel Inspection Program

In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected at least once every 10 years. This inspection shall be by either of the following examinations:

a. An in-place examination, utilizing ultrasonic testing, over the volume from the inner bore of the flywheel to the circle of one-half the outer radius; or

b. A surface examination, utilizing magnetic particle testing and/or penetrant testing, of the exposed surfaces of the disassembled flywheel.

INSERT 2

Attachment 4

Mark-up of Proposed Technical Specification Bases Changes

**INSERT 3**

in accordance with the Surveillance Frequency Control Program

**INSERT 4**

The surveillance frequency is controlled under the Surveillance Frequency Control Program.

## 3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120, and 228 steps withdrawn for the Control Banks and 18, 210, and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with rods at their individual mechanical fully withdrawn position, $T_{avg}$ greater than or equal to 551°F and 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.

The fully withdrawn position of shutdown and control banks can be varied between 225 and the mechanical fully withdrawn position (up to 232 steps), inclusive. An engineering evaluation was performed to allow operation to the 232 step maximum. The 225 to 232 step interval allows axial repositioning to minimize RCCA wear.

INSERT 3

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 LCOs are satisfied.

For Specification 3.1.3.1 ACTIONS b. and c., it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). Trippability is defined in Attachment C to a letter dated December 21, 1984, from E. P. Rahe (Westinghouse) to C. 0. Thomas (NRC). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus falls under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately 4 hours for this verification.

## 3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA, the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

INSERT - 3

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

$F_{\Delta H}^{N}$ will be maintained within its limits provided Conditions a. through d. above are maintained. Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. There is additional margin available to offset any other DNBR penalties and for plant design flexibility.

When an $F_Q(Z)$ measurement is taken, an allowance for both measurement error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the movable incore detectors, while 5.21% is appropriate for surveillance results determined with the fixed incore detectors. A 3% allowance is appropriate for manufacturing tolerance.

INSERT 3

The hot channel factor $F_Q^M(Z)$ is measured ~~periodically~~ and increased by a cycle and height dependent power factor appropriate to Relaxed Axial Offset Control (RAOC) operation, W(Z), to provide assurance that the limit on the hot channel factor $F_Q(Z)$ is met. W(Z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(Z) function for normal operation is specified in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6.

When RCS $F_{\Delta H}^{N}$ is measured, no additional allowances are necessary prior to comparison with the established limit. Appropriate $F_{\Delta H}^{N}$ measurement uncertainties are already incorporated into the limits $F_{\Delta H}^{N}$ established in the CORE OPERATING LIMITS REPORT for each measurement system, and a bounding $F_{\Delta H}^{N}$ measurement uncertainty has been applied in determination of the design DNBR value. The appropriate $F_{\Delta H}^{N}$ measurement uncertainties are 4.13% for the fixed incore detector system and 4% for the movable incore detector system.

3/4.2.4 QUADRANT POWER TILT RATIO

The purpose of this specification is to detect gross changes in core power distribution between monthly Incore Detector System surveillances. During normal operation the QUADRANT POWER TILT RATIO is set equal to 1.0 once acceptability of core peaking factors has been established by review of incore surveillances. The limit of 1.02 is established as an indication that the power distribution has changed enough to warrant further investigation.

## 3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the updated FSAR assumptions and have been analytically demonstrated adequate to assure compliance with acceptance criteria for each analyzed transient. Operating procedures include allowances for measurement and indication uncertainty so that the limits specified in the COLR for $T_{avg}$ and for pressurizer pressure are not exceeded.

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.

INSERT 4

The periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the specified limit.

RCS flow must be greater than or equal to, 1) the Thermal Design Flow (TDF) with an allowance for measurement uncertainty and, 2) the minimum measured flow used in place of the TDF in the analysis of the DNB related events when the Revised Thermal Design Procedure (RTDP) methodology is utilized. Measurement of RCS total flow rate is performed by performance of either a precision calorimetric heat balance or normalized cold leg elbow tap $\Delta P$ measurements. RCS flow measurements using either the precision heat balance or the elbow tap $\Delta P$ measurement methods are to be performed at steady state conditions prior to operation above 95% rated thermal power (RTP) at the beginning of a new fuel cycle. The elbow tap RCS flow measurement methodology is described in WCAP-15404, "Justification of Elbow Taps for RCS Flow Verification at Seabrook Station", dated April 2000.

## 3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed ~~at the minimum frequencies~~ are sufficient to demonstrate this capability.

INSERT 3

Table 3.3-1 contains the action statements for inoperable Reactor Trip System Instrumentation. Actions 4 and 5, associated with the source range neutron flux instruments, each include a requirement to suspend operations involving positive reactivity changes. When complying with this action, operations that individually add limited, positive reactivity are acceptable when, combined with other actions that add negative reactivity, the overall net reactivity addition is zero or negative. For example, a positive reactivity addition caused by temperature fluctuations from inventory addition or temperature control fluctuations is acceptable if it is combined with a negative reactivity addition such that the overall, net reactivity addition is zero or negative.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. The NRC Safety Evaluation Reports for WCAP-10271 and its supplements and revisions were provided on February 21, 1985, February 22, 1989 and April 30, 1990.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (continued)

uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R\ S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span; R or Rack Error is the "as measured" deviation, in the percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

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

INSERT 3

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.).

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, signal processing and actuation logic 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.

## MONITORING INSTRUMENTATION

### 3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS (Continued)

and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

### 3/4.3.3.2 (THIS SPECIFICATION NUMBER IS NOT USED)

### 3/4.3.3.3 (THIS SPECIFICATION NUMBER IS NOT USED)

### 3/4.3.3.4 (THIS SPECIFICATION NUMBER IS NOT USED)

### 3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of Appendix A to 10 CFR Part 50.

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, control, and power circuits and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10CFRPart 50.

INSERT 1

The Technical Specifications (T/S) require surveillance testing of selected equipment used for safe shutdown from outside the control room at Remote Safe Shutdown (RSS) locations. The required equipment is listed in Table 3.3-9. The selection criteria for the Transfer Switch/Control Circuit portion of the table is the primary equipment which has remote/local selector switches and is required to perform the reactor coolant system inventory and pressure control, reactivity control, and decay heat removal functions to achieve and maintain hot standby. Redundant, safety grade equipment is provided for GDC 19 shutdown. For Appendix R shutdown, only one train of equipment (safety or non-safety related) is required; redundancy is not a requirement. Therefore, some equipment in Table 3.3-9 is required for a GDC 19 shutdown but not for a GDC 3/Appendix R shutdown. Seabrook is a hot standby safe shutdown design basis plant (see UFSAR Section 5.4.7.2.i). Support equipment, and equipment required only to achieve and maintain cold shutdown, are not required to be included in the T/S table.

## 3/4.4.3 PRESSURIZER

INSERT 4

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 Reactor Coolant System pressure and establish natural circulation.

## 3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The PORVs and their associated block valves are powered from Class 1E power supply busses.

The PORVs are equipped with automatic actuation circuitry and manual control capability. The PORVs are considered OPERABLE in either the automatic or manual mode for the following reasons:

(1) No credit is taken in any FSAR accident analysis for automatic PORV actuation to mitigate the consequences of an accident.

(2) No Surveillance Requirement (ACOT or TADOT) exists for verifying automatic operation.

(3) The required ACTION for an inoperable PORV(s) (closing the block valve) conflicts with any presumed requirement for automatic actuation.

## REACTOR COOLANT SYSTEM LEAKAGE

### 3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

### 3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

ACTIONS (c) (Continued)

The 12 hour interval is sufficient to detect increasing RCS leakage. The Action provides 7 days to restore another RCS leakage monitor to operable status to regain the intended leakage detection diversity. The 7 day restoration time ensures that the plant will not be operated in a degraded configuration for a lengthy time period. Two leakage detections systems must be restored to operable status within 30 days to meet the LCO or the plant must shutdown.

SURVEILLANCE REQUIREMENTS

INSERT 4

SR 4.4.6.1.a.1

SR 4.4.6.1.a.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 4.4.6.1.a.2

SR 4.4.6.1.a.2 requires the performance of a digital channel operational test on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 4.4.6.1.a.3 and 4.4.6.1.b

These SRs require the performance of a channel calibration for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.
3. FSAR, Section 5.2.5.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

SURVEILLANCE REQUIREMENTS

4.4.6.2.1

Verifying RCS leakage to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. Unidentified leakage and identified leakage are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The surveillance is modified by two footnotes. Footnote 1 states that this SR is not applicable to primary to secondary leakage because leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance. Footnote 2 states that this SR 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 RCP seal injection and return flows.

An early warning of pressure boundary leakage or unidentified leakage is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. These leakage detection systems are specified in LCO 3.4.6.1, "RCS Leakage Detection Instrumentation."

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 4

## REACTOR COOLANT SYSTEM LEAKAGE

### 3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

SR 4.4.6.2.1.f verifies that primary to secondary leakage is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this SR 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 as described in Reference 5. The operational leakage rate limit applies to leakage through any one SG. If it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG.

The Surveillance is modified by a footnote that states 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 RCP seal injection and return flows.

INSERT 4

~~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 the EPRI guidelines (Ref. 5).

### 4.4.6.2.2

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. RCS Pressure Isolation Valve (PIV) Leakage measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS leakage when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE.

### REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI, "Pressurized Water Reactor Primary-to Secondary Leak Guidelines."

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (Continued)

An automatic valve may be aligned in other than its accident position provided (1) the valve receives an automatic signal to re-position to its required position in the event of an accident, and (2) the valve is otherwise operable (stroke time within limits, motive force available to re-position the valve, control circuitry energized, and mechanically capable of re-positioning).

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, non-operating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the refueling water storage tank (RWST) and from the ECCS recirculation sump to the RCS full of water (by verifying at the accessible ECCS piping high points and pump casings, excluding the operating centrifugal charging pump) ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of non-condensable gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following a safety injection (SI) signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

INSERT 4

It should be noted that Surveillance Requirement 4.5.2b.1 Bases also

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (Continued)

conditions the Surveillance Requirement by stating that verification is to be performed at the "accessible" ECCS piping high points and pump casing, excluding the operating centrifugal charging pump. Thus, the Bases recognizes that certain "impracticalities," i.e., physical accessibility issues or the operating centrifugal charging pump (only) under dynamic conditions, may preclude verification at certain points and as such provides relief. However, such relief cannot be taken at the expense of possible system inoperability because of lack of periodic verification. Such relief can only be taken if there is reasonable assurance that the collection of gasses or void formation is of no significant concern at the points not to be verified periodically within the ~~stipulated~~ surveillance interval ~~(i.e., every 31 days)~~. Furthermore, because of regulatory requirements, even if reasonable assurance can be justified for not requiring verification at a particular high point, such verification must be performed if the high point is accessible. "Inaccessibility" cannot be used as a mere convenience.

ECCS piping high points may be considered inaccessible if any of the following criteria are met:

a) The high point is located inside the bioshield in containment while the reactor is critical (Modes 1 & 2), since this area can contain lethal radiation fields during reactor operation. During those situations when the reactor is not critical, other conditions where gaining access poses a safety or radiological hazard (e.g., high system temperature, high radiological conditions) may prohibit verification by UT/venting.

b) The high point is located in an area where gaining access poses a safety or radiological hazard, e.g.:

- Installation/removal of temporary ladders within containment or other areas where stay times (heat stress / high radiation levels) or other factors must be kept to minimums.

Note: The safety or radiological concern should be documented for further evaluation by the responsible organization(s).

c) High points within heat exchanger tubes.

The phrase "full of water" is subjective particularly since most system fluid streams do contain a certain amount of non-condensable gasses. ECCS piping may be considered "full of water" if there is reasonable assurance that the content of the non-condensable gas within the system (including the aggregate amount of non-condensable gasses in all ECCS piping) and at a particular point will not be of significance to impair the ECCS system from performing

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM (Continued)

ACTIONS

Note 1 prohibits the application of LCO 3.0.4.b to an inoperable EFW train when entering MODE 1, and Note 2 prohibits the application of LCO 3.0.4.b to an inoperable startup feedwater pump. There is an increased risk associated with entering MODE 1 with AFW inoperable, or entering MODES 3 or 2 with the startup feedwater pump inoperable. The provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

With one AFW pump inoperable, the action provides a 72-hour AOT for restoring the pump to an operable status before requiring a plant shutdown. This time is reasonable based on the availability of redundant equipment and the low probability of an accident occurring during this time. Additional actions with more limiting AOTs apply to conditions involving more than one inoperable AFW pump. In the event that all AFW pumps are inoperable, the plant is in a seriously degraded condition. Consequently, the plant should not be perturbed by any action, including a power change, that might result in a plant trip and demand on the EFW system. The seriousness of this condition requires immediately initiating corrective action to restore at least one AFW pump to operable status as soon as possible.

INSERT 3

SURVEILLANCES

Various surveillance requirements, with frequencies ranging ~~from 31~~ days ~~to eighteen months~~, demonstrate the operability of the AFW system. ~~Every 31 days~~, each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured, is verified in its correct position. This verification includes only those valves in the direct flow path through safety-related equipment whose position is critical to the proper functioning of the safety-related equipment. Vents, drains, sampling connections, instrument taps, etc., that are not directly in the flow path and are not critical to proper functioning of the safety-related equipment are excluded from this surveillance requirement.

Testing of the steam-driven EFW pump is exempt from the provisions of TS 4.0.4 for entry into MODE 3. This allowance is necessary because the surveillance testing, which requires a minimum steam pressure of 500 psig, cannot be performed until the plant reaches MODE 3. Once steam pressure reaches 500 psig, administrative controls establish a 24-hour time limit for completing the testing consistent with Specification 4.0.4.

## 3/4.7.1 TURBINE CYCLE (Continued)

### 3/4.7.1.4 SPECIFIC ACTIVITY (Continued)

SURVEILLANCE REQUIREMENTS (SR)

SR 4.7.1.4

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31-day frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT ~~I-131~~, and allows ~~for~~ appropriate action ~~to be~~ taken to maintain ~~levels~~ below the ~~LCO~~ limit.

INSERT 4

REFERENCES

1. 10 CFR 50.67.
2. UFSAR, Chapter 15.

### 3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

### 3/4.7.1.6 ATMOSPHERIC RELIEF VALVES

The OPERABILITY of the Atmospheric Relief Valves (ARVs) ensures the controlled removal of reactor decay heat during reactor cooldown, plant startup, and after a turbine trip, when the condenser and/or the turbine bypass system are not available. When available, the ARVs can be used to reduce main steam pressure for both hot shutdown and cold shutdown conditions. The ARVs provide a method for cooling the plant to residual heat removal entry conditions should the turbine bypass system to the condenser be unavailable. This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST).

3/4.7.4 SERVICE WATER SYSTEM/ULTIMATE HEAT SINK (Continued)

The portable makeup pump must have a minimum capacity of 200 gpm, which ensures the capability to meet the calculated makeup requirement of 140 gpm at seven days after a LOCA. A surveillance requirement verifies the ability of the pump to produce flow of at least 200 gpm ~~every 18 months~~. In addition, ~~a monthly~~ inventory and periodic inspections of the hose confirm the availability and integrity of sufficient flexible hose.
an

INSERT 3

The seven-day period during which the cooling tower can operate without makeup water provides adequate time to move the portable pump into position, lay the hose, and make the system ready for operation. As a result, the portable pump is not necessarily immediately available for operation when stored in its design operational readiness state. The seven-day period allows ample time to charge the battery, obtain diesel fuel, inflate the trailer tires, and obtain a tow vehicle.

Switchover from the service water pumphouse to the mechanical draft cooling tower is accomplished either automatically (Tower Actuation (TA) signal) or manually. Manual action is required to realign the system from the cooling tower to the service water pumphouse. While a cooling tower pump is operating, interlocks prevent the train associated service water pumps from starting. To provide additional protection, during operation while aligned to the cooling tower, the service water pump control switches may be maintained in the pull-to-lock position to prevent inadvertent pump operation. As previously discussed, realignment to the service water pumphouse requires manual action; maintaining the control switches in the pull-to-lock position does not change this required action sequence. Pump operation is not affected by maintaining the control switches in the pull-to-lock position during this period; therefore, OPERABILITY of the service water pumps is not compromised.

The limitations on service water pumphouse minimum water level and the requirements for cooling tower OPERABILITY are based on providing a 30-day cooling water supply to safety-related equipment without exceeding the safety related equipment design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

The Cooling Tower is normally aligned to allow return flow to bypass the tower sprays 4 and return to the basin. Upon receipt of a Tower Actuation Signal, the fans and sprays are manually operated as required. This manual operation, which is governed by procedures, ensures that ice does not buildup on the cooling tower tile fill and fans. The cooling tower basin temperature limit of 70°F provides sufficient time for manual initiation of the cooling tower sprays and fans following the design basis seismic event with a concurrent LOCA, during the design extreme ambient temperature conditions. Under this scenario, manual action is sufficient to maintain the cooling tower basin at a temperature which precludes equipment damage during the postulated design basis event.

3/4.7.6 CONTROL ROOM SUBSYSTEMS (Continued)

SURVEILLANCE REQUIREMENTS

SR 4.7.6.1

Periodic

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. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. Systems with heaters must be operated for ≥10 continuous hours with the heaters energized. ~~The 31-day Frequency is based on the reliability of the equipment and the two-train redundancy.~~

INSERT 3

SRs also periodically test the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the ~~activated charcoal~~.

INSERT 4

The SRs verify that each CREMAFS train starts and operates on test actuation signals. ~~The Frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle.~~

SR 4.7.6.2

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Action b. must be entered. Action b.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3 (Ref. 5), which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). These compensatory measures may also be used as mitigating actions as required by Action b.2. Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

3/4.7.6 CONTROL ROOM SUBSYSTEMS (Continued)

REFERENCES

1. FSAR, Section 6.4
2. FSAR, Chapter 15
3. FSAR, Section 6.4.4.2
4. FSAR, Section 6.4
5. Regulatory Guide 1.196
6. NEI 99-03, "Control Room Habitability Assessment"

AIR CONDITIONING

The OPERABILITY of the safety-related Control Room Air Conditioning Subsystem ensures that the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system is not exceeded. The safety-related Control Room Air Conditioning Subsystem consists of two independent and redundant trains that provide cooling of recirculated control room air. The design basis of the safety-related Control Room Air Conditioning Subsystem is to maintain the control room temperature for 30 days of continued occupancy. The safety-related chillers are designed to operate in conditions down to the design basis winter temperature. When the chiller units unload due to insufficient heat load on the system, each Control Room air Conditioning Subsystem remains operable. Surveillance to demonstrate OPERABILITY will verify each subsystem has the capability to maintain the control room area temperature less than the limiting equipment qualification temperature. The operational surveillance will be performed ~~on a quarterly basis~~, requiring each safety-related Control Room Air Conditioning Subsystem to operate over a twenty-four hour period. This will ensure the safety related subsystem can remove the heat load based on daily cyclic outdoor air temperature.

INSERT 3

The Control Room Air Conditioning fans are necessary to support both the operation of the Control Room Emergency Makeup Air and Filtration and the Control Room Air Conditioning Subsystems.

3/4.8.1 AC SOURCES (Continued)

**LIMITING CONDITION FOR OPERATION (LCO) (continued)**

**f.** With Train A and Train B EDGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. For this level of degradation, the offsite electrical power system is the only source of AC power available. The risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability and inadvertent generator trip, which could result in a total loss of AC power); however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Reference 6, with both EDGs inoperable, operation may continue for a period that should not exceed 2 hours. If one EDG is restored within 2 hours power operation may continue in accordance with ACTION b.

Following the 2-hour AOT, ACTION f. requires that both diesel generators be restored to Operable status within 72 hours. The requirement for restoring both diesel generators to OPERABLE status within 72 hours may be extended to 14 days to perform either extended preplanned maintenance (both preventive and corrective) or extended unplanned corrective maintenance work. Prior to exceeding the 72-hour AOT the SEPS must be available and an operational readiness status check performed in accordance with Technical Requirement (TR) 31. Refer to Bases for ACTION b. for additional information and requirements.

**SURVEILLANCE REQUIREMENTS (SR)**

INSERT 3

The AC sources are designed to permit inspection and testing of important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref 8). Periodic component tests are supplemented by extensive functional tests ~~during refueling outages~~ (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the EDGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 9), and Regulatory Guide 1.137 (Ref. 10), as addressed in the UFSAR including exceptions thereto.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of 3740 Vac is 90% of the nominal 4160 Vac output voltage. This value, which is specified in ANSI C84.1 (Ref 11) allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 Vac. It also allows for voltage drops to motors and other equipment down through the 120 Vac level where minimum operating voltage is also usually specified as 90% of nameplate rating. The specified maximum steady state output voltage of 4580 Vac is equal to the nominal bus voltage plus 10%. The specified minimum and maximum frequencies of the EDG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ±2% of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3).

**SR 4.8.1.1.1a**

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. ~~The 7-day frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.~~



INSERT 4

3/4.8.1 AC SOURCES (Continued)

**SURVEILLANCE REQUIREMENTS (SR) (continued)**

**SR 4.1.1.1b**

INSERT 4

Transfer of each 4.16 kV ESF bus power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The transfer circuit is only required to be OPERABLE when the offsite circuit to which it transfers is credited as being OPERABLE. The 18-month frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18-month frequency. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.

SR 4.8.1.1.1.b is modified by footnote * prohibiting performance during MODE 1 or 2. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems.

**SR 4.8.1.1.2a through 2.g**

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

SR 4.8.1.1.2 is modified by footnote ** to indicate that all planned EDG starts for the purposes of these surveillances may be preceded by an engine prelube period. This allowance is to minimize wear on moving parts since the EDG does not get lubricated when the engine is not running.

The term “standby condition” used throughout these SRs mean that the diesel engine coolant and oil are being continuously circulated and engine temperature is being maintained consistent with manufacturer recommendations at keep-warm values.

INSERT 3

**SR 4.8.1.1.2a**

Activities to demonstrate EDG OPERABILITY under this SR are to be performed ~~on a STAGGERED TEST BASIS at least once every 31 days. Performance of surveillances on a staggered test basis provides an added measure of assurance that the redundant onsite power sources are OPERABLE and any detected failure during surveillance testing is promptly evaluated to determine if the failure has a common failure mode component to it.~~

**SR 4.8.1.1.2a.1)** provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 hour of EDG operation at full load plus 10%. ~~The 31-day frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and facility operators would be aware of any large uses of fuel oil during this period.~~

**SR 4.8.1.1.2a.2)** provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each EDG's operation for 7 days. The 7-day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location. ~~The 31-day frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.~~

INSERT 4

3/4.8.1 AC SOURCES (Continued)

INSERT 4

**SURVEILLANCE REQUIREMENTS (SR) (continued)**

**SR 4.8.1.1.2a.3)** demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE. The ~~31-day~~ frequency is ~~appropriate~~ since proper ~~operation~~ of fuel ~~transfer~~ systems ~~is an~~ inherent part of EDG OPERABILITY.

**SR 4.8.1.1.2a.4)** ensures that sufficient lube oil inventory is available to support at least 7 days of operation for each EDG. The 275 gal minimum requirement is based on the EDG manufacturer consumption values for the run time of the EDG. Implicit in this SR is the requirement to verify the capability to transfer the lube oil from its storage location to the EDG, when the EDG lube oil sump does not hold adequate inventory for 7 days of operation without the level reaching the manufacturer recommended minimum level. A 31-day ~~frequency~~ is ~~adequate~~ to ensure ~~that a sufficient~~ lube oil ~~supply is~~ onsite, since ~~EDG~~ starts ~~and run~~ time are ~~closely~~ ~~monitored by~~ the unit staff.

**SR 4.8.1.1.2a.5)** ensures that the EDG is capable of starting from standby conditions and attaining rated voltage and frequency. Footnote *** allows a modified start procedure to be used in lieu of the 10-12 seconds "fast start" for the EDG. In order to reduce stress and wear on diesel engines, the manufacturer recommends a modified start in which the starting speed of the EDG is limited, warmup is limited to this lower speed, and the EDG is gradually accelerated to synchronous speed prior to loading. Use of the modified start method requires the diesel governor system to be capable of engine idling and gradual acceleration to synchronous speed. When the modified start is not used footnote *** requires that the time, voltage, and frequency tolerances of SR 4.8.1.1.2e) (10 second start) be met. The 31-day ~~frequency~~ for SR ~~4.8.1~~.1.2a.5) is ~~consistent~~ with Regulatory ~~Guide~~ 1.9 (Ref. 3), ~~though~~ Seabrook Station is ~~not committed~~ to ~~Regulatory Guide~~ 1.9.

**SR 4.8.1.1.2a.6)** verifies that the EDG is capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the EDG is connected to the offsite source.

To minimize mechanical stress and wear on the diesel engine SR 4.8.1.1.2a.6) is modified by footnote **** that allows EDG loading per the manufacturers recommendations, including a warmup period. In addition, footnote **** states that momentary transients outside the load range, due to changing bus conditions do not invalidate the test. Footnote **** also stipulates a prerequisite requirement for performance of this SR whereby this SR must be preceded by and immediately follow a successful EDG start per SR 4.8.1.1.2a.5) or SR 4.8.1.1.2e to credit satisfactory performance.

Note that although no power factor requirements are established by SR 4.8.1.1.2a.6), the EDG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the EDG. Routine overloading may result in more frequent tear down inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY. Similarly, though not stated in footnote ****, momentary kvar transients above the limit do not invalidate the test.

The ~~31-day~~ ~~frequency for~~ SR 4.~~8.1.1.2~~a.6) is ~~consistent~~ ~~with Regulatory Guide 1.9~~ (~~Ref. 3~~).



INSERT 4

3/4.8.1 AC SOURCES (Continued)

**SURVEILLANCE REQUIREMENTS (SR) (continued)**

**SR 4.8.1.1.2a.7)** ensures that following EDG testing per SR 4.8.1.1.2a.5) and SR 4.8.1.1.2a.6) that the EDG is returned to ready to standby status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the EDG to reload if a subsequent loss of offsite power occurs. The EDG is considered to be in ready to load status when the EDG is at rated speed and voltage, the output breaker is open and can receive an autoclose signal on bus undervoltage, and the load sequence timers are reset.

**SR 4.8.1.1.2b and SR 4.8.1.1.2c**

INSERT 3

Removal of water from the fuel oil day and storage tanks ~~once every 31 days~~ eliminates the necessary environment for bacterial survival. Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during EDG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. ~~The Surveillance frequencies are established by Regulatory Guide 1.137 (Ref. 10).~~ INSERT 4 This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.

**SR 4.8.1.1.2d**

For proper operation of the standby EDGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 10) addresses the recommended fuel oil practices as supplemented by ANSI Standards. The SR refers to the Diesel Fuel Oil Testing Program (Specification 6.7.6i) for the verification of new and stored fuel oil properties. The fuel oil properties governed by Specification 6.7.6i are water and sediment content, kinematic viscosity, specific gravity (or API gravity), and impurity level. Technical Requirements Program (TRP) 5.1 implements the requirements of Specification 6.7.6i. ~~The 31-day frequency is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on EDG operation.~~ This Surveillance ensures the availability of high quality fuel oil for the EDGs.

**SR 4.8.1.1.2e**

INSERT 3

This surveillance requires that, ~~at a 184-day frequency~~ the EDG starts from standby conditions and achieves required voltage and frequency within 10 seconds (a.k.a, "fast start"). The 10-second start requirement supports the assumptions of the design basis LOCA analysis in the UFSAR, Chapter 15 (Ref. 5).

Upper limits for voltage and frequency are not specified during the initial EDG start in order to account for potential overshoot in voltage and frequency because of governor control system characteristics when testing the EDG in an unloaded condition.

Since this SR requires a 10 second start, it is more restrictive than SR 4.8.1.1.2a.5), and it may be performed in lieu of SR 4.8.1.1.2a.5). Associated footnote # allows crediting of this SR for SR 4.8.1.1.2a.5). Additionally, footnote # stipulates that gradual loading per SR 4.8.1.1.2a.6) must immediately follow this surveillance.

In addition to the SR requirements, the time for the EDG to reach steady state operation, unless the modified EDG start method is employed, is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

3/4.8.1 AC SOURCES (Continued)

**SURVEILLANCE REQUIREMENTS (SR) (continued)**

This SR in combination with SR 4.8.1.1.2a.5) help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition. INSERT 4

The 184-day frequency is consistent with Generic Letter 84-15 (Ref. 7) and provides adequate assurance of EDG OPERABILITY, while minimizing degradation resulting from testing.

**SR 4.8.1.1.2f**

INSERT 3

Surveillances carried out under SR 4.8.1.1.2f are activities normally conducted during shutdown at a refueling frequency of every 18 months. The SR is modified by footnote ## which provides a dispensation from the 'during shutdown' requirement provided an evaluation supports the safe conduct of a particular surveillance in a condition or mode that is consistent with safe operation of the plant. This disposition is consistent with Generic Letter 91-04 (Ref. 13).

Note: SR 4.8.1.1.2f.1) and SR 4.8.1.1.2.2f.13) are Not Used.

**SR 4.8.1.1.2f.2)** demonstrates the EDG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency limits. This surveillance may be accomplished by either:

a. Tripping the EDG output breaker with the EDG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus, or

b. Tripping its associated single largest post-accident load with the EDG solely supplying the bus.

If method a. is used the EDG power factor must be in the range of 0.9 which is representative of actual design basis inductive loading.

The voltage and frequency specified are consistent with the design range of the equipment powered by the EDG and are the steady state voltage and frequency values to which the system must recover following load rejection. The 18-month frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9). INSERT 4

**SR 4.8.1.1.2f.3)** demonstrates the DG's capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG does not trip upon loss of the load. These acceptance criteria provide for DG damage protection. While the DG is not expected to experience this transient during an event and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

As required by IEEE-387 (Ref. 12), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.

The 18 month frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9) and is intended to be consistent with expected fuel cycle lengths.

3/4.8.1 AC SOURCES (Continued)

**SURVEILLANCE REQUIREMENTS (SR) (continued)**

**SR 4.8.1.1.2f.4)** demonstrates the as designed operation of the standby power sources during loss of the offsite source, as required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1). This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the EDG. It further demonstrates the capability of the EDG to automatically achieve the required voltage and frequency within the specified time.

The EDG auto-start time of 12 seconds is derived from requirements of the accident analysis to respond to a loss of offsite power event. The Surveillance must be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability is achieved.

The requirement to verify the connection and power supply of permanent and auto-connected shutdown loads is intended to satisfactorily show the relationship of these loads to the EDG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. In lieu of actual demonstration of connection and loading of loads, testing and analysis that adequately show the capability of the EDG systems to perform these loading functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified. Similarly, pumps need not be operated at design basis flows since the purpose of the SR is only to verify correct loading sequence.

This SR is modified by footnote ### to allow starting of the diesel engine at or near normal operating temperature in lieu of standby conditions. The reason for the footnote is to minimize wear and tear on the EDGs during testing. Repeated fast starts with the diesel engine starting at a standby condition temperature still contribute to accelerated engine degradation. Starting of the diesel generator from standby conditions, equivalent to the keep-warm systems temperature, would continue to be performed per SR 4.8.1.1.2f.6) (the loss-of-offsite power in conjunction with a SI actuation test signal) which would meet the spirit of Generic Letter 84-15 (Ref. 7). This allowance would also benefit outage planning and scheduling to shorten the length of the outage by not needing to wait for the engine to cool down before starting the next test. In addition, this capability would continue to be verified ~~several times~~ during the 18-~~month~~ oper~~ating~~ cycle ~~when performing the 184-day~~ fast start test per SR 4.8.1.1.2e.

The 18-month frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

INSERT 4

**SR 4.8.1.1.2f.5)** demonstrates that the EDG automatically starts and reaches the minimum voltage and frequency requirements within the specified time (10 seconds) from the design basis (LOCA) actuation signal (SI signal) without loss of offsite power, maintains steady-state voltage and frequency within prescribed limits, and operates on standby for at least 5 minutes. The 5-minute period provides sufficient time to demonstrate stability. Upper limits for voltage and frequency are not specified during the initial EDG start in order to account for potential overshoot in voltage and frequency because of governor control system characteristics when testing the EDG in an unloaded condition. The time, voltage and frequency for the EDG to reach steady state operation is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The 18-month frequency takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18-month frequency. The frequency is also consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9) for other EDG surveillance requirements. Therefore, the frequency was concluded to be

3/4.8.1 AC SOURCES (Continued)

**SURVEILLANCE REQUIREMENTS (SR) (continued)**

acceptable ~~from a~~ reliability ~~standpoint.~~

This SR is modified by footnote ###, as described in SR 4.8.1.1.2f.4), to minimize wear and tear on the EDGs during testing.

**SR 4.8.1.1.2f.6)** demonstrates the EDG operation, as discussed in the Bases for SR 4.8.1.1.2f.4), during a loss of offsite power actuation test signal in conjunction with a SI actuation signal. In the event of a DBA coincident with a loss of offsite power, the EDGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. The basis for the EDG auto-start is as discussed in the Bases for SR 4.8.1.1.2f.5). The basis for the EDG loading is as discussed in the Bases for SR 4.8.1.1.2f.4).

The surveillance must be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability is achieved.

The SR is performed with the EDG initially at standby condition, i.e., equivalent to the keep-warm systems temperature. This requirement is consistent with Generic Letter 84-15 (Ref. 7) which notes that the design basis for the plant, i.e., large LOCA coincident with loss of offsite power requires the EDG to be capable of starting from ambient conditions (keep-warm system temperature).

The SR also demonstrates that all automatic protective trip functions (e.g., high jacket water temperature) except, engine overspeed, 4160 volt bus fault, generator differential current, and low lube oil pressure, are bypassed on a loss of voltage signal concurrent with a SI actuation test signal. The noncritical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The EDG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the EDG.

INSERT 4

~~The 18-month frequency takes into consideration unit conditions required to perform the Surveillance, is intended to be consistent with an expected fuel cycle length of 18 months, and is consistent with the frequency of SR 4.8.1.1.2f.4). Operating experience has shown that these components usually pass the SR when performed at the 18-month frequency. The frequency is also consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9) for other EDG surveillance requirements. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.~~

**SR 4.8.1.1.2f.7)** demonstrates that the EDGs can start and run continuously at full load capability for an interval of not less than 24 hours at a load equivalent to 92 - 100 percent of the continuous duty rating of the EDG. The EDG starts for this SR can be performed either from standby or hot conditions. The load band is provided to avoid routine overloading of the EDG. Routine overloading may result in more frequent tear down inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

Should auto-connected loads be added in the future such that the load on the bus reach or exceed the EDG continuous load rating, the EDG must run for a minimum of 2 hours at a load equivalent to 105 – 110 percent the continuous duty rating of the EDG. The remaining hours of the 24-hour run are to be at 92 – 100 percent full load. In addition, the SR requires verification that the auto-connected loads do not exceed the short term rating of the EDG.

Note that although no power factor requirements are established by SR 4.8.1.1.2f.7), the EDG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the EDG. Routine

3/4.8.1 AC SOURCES (Continued)

**SURVEILLANCE REQUIREMENTS (SR) (continued)**

overloading may result in more frequent tear down inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

To minimize mechanical stress and wear on the diesel engine SR 4.8.1.1.2f.7) is modified by footnote #### that allows EDG loading per the manufacturers recommendations, including a warmup period. In addition, the footnote states that momentary transients outside the load range, due to changing bus conditions do not invalidate the test. Similarly, though not stated in footnote ####, momentary kvar transients above the limit do not invalidate the test.

INSERT 4

The 18-month frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref 9), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

**SR 4.8.1.1.2f.8)** demonstrates that the diesel engine can restart within 5 minutes from a hot condition, such as subsequent to shutdown from normal surveillances, and achieve the minimum required voltage and frequency within 10 seconds and steady-state conditions thereafter. The time, voltage and frequency for the EDG to reach steady state operation is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The requirement that the diesel has operated for at least 2 hours at sufficiently loaded conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. The load band is provided to avoid routine overloading of the EDG.

The SR is modified by footnote + noting that momentary transients outside the load range, due to changing bus loads, do not invalidate the test.

The 18-month frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(5).

**SR 4.8.1.1.2f.9)** ensures, as recommended by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), that the manual synchronization and load transfer (emergency loads) from the EDG to the offsite source can be made and the EDG can be returned to standby status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the EDG to reload if a subsequent loss of offsite power occurs. The EDG is considered to be in standby status when the EDG is aligned for auto-start, the EDG circuit breaker is available for automatic closure, and the emergency power sequencer timer(s) are reset and available for automatic operation.

The three sub-steps do not need to be performed sequentially. It is acceptable to delay performance of sub-step c) to support optimum scheduling of maintenance and surveillance activities so long as the requisite test criteria are met when it is performed.

The 18-month frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.

**SR 4.8.1.1.2f.10)** is a demonstration of the test mode override which ensures that EDG availability under accident conditions will not be compromised as a result of testing the EDG while connected to its bus. The EDG is verified to return to standby operation and the emergency loads are automatically energized with offsite power if a SI actuation signal is received during operation in the test mode. Ready to load operation is defined as the EDG running at rated speed and voltage with the EDG output breaker open.

The requirement to automatically energize the emergency loads with offsite power is intended

3/4.8.1 AC SOURCES (Continued)

INSERT 4

**SURVEILLANCE REQUIREMENTS (SR) (continued)**

to show that the emergency loading was not affected by the EDG operation in test mode. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The 18-month frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(8), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

**SR 4.8.1.1.2f.11)** demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from each storage tank to each EDG day tank via the installed cross-connection lines. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for fuel transfer systems are OPERABLE.

The 18-month frequency for this SR is appropriate to verify this capability since both EDG and associated fuel oil trains are independent and are normally not cross-connected. The cross-connect provision is an installed feature for an enhanced defense-in-depth capability should, in the unlikely event, it become necessary to cross-connect the fuel oil trains.

**SR 4.8.1.1.2f.12)** ensures that under loss of offsite power conditions, with or without an accident, loads are sequentially connected to the bus by the emergency power sequencer timer. The sequencing logic controls the permissive and starting signals to motor and other load breakers to prevent overloading of the EDGs due to high inrush starting currents. The 10% load sequence time interval tolerance ensures that sufficient time exists for the EDG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

The 18-month frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(2), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

**SR 4.8.1.1.2f.14)** demonstrates that when a Tower Actuation (TA) signal is generated, while the EDG is loaded with its permanently connected loads and auto-connected emergency accident loads, the associated operating service water pump automatically trips and the corresponding cooling tower pump starts and after energization that voltage and frequency of the emergency bus remains within steady-state limits.

The 18-month frequency takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18-month frequency. The frequency is also consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9) for other EDG surveillance requirements. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.

3/4.8.1 AC SOURCES (Continued)

**SURVEILLANCE REQUIREMENTS (SR) (continued)**

**SR 4.8.1.1.2f.15)** demonstrates that while EDG 1A is loaded with its permanently connected loads and auto-connected emergency loads, that emergency bus E5 voltage and frequency remain within steady-state limits after manual energization of the 1500 hp startup feedwater pump (the largest manually-connected load).

The 18-month frequency takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18-month frequency. The frequency is also consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9) for other EDG surveillance requirements. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.

INSERT 4

**SR 4.8.1.1.2g**

This surveillance demonstrates that the EDG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper voltage and frequency within 10 seconds then steady-state condition when the EDGs are started simultaneously. The time, voltage and frequency for the EDG to reach steady state operation is monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The SR also requires that the EDGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations at keep-warm values.

The 10-year frequency is consistent with the recommendations of RG 1.108 (Ref. 9).

MODES 5 AND 6

During operation in MODEs 5 and 6, the required AC sources include one off-site circuit capable of supplying the on-site Class 1E distribution system and an operable emergency diesel generator. These minimum AC sources ensure that (1) the unit can be maintained in the shutdown condition, (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit, and (3) adequate AC power is available to mitigate an event postulated to occur during shutdown.

If the minimum required AC sources are not operable, the action statement requires immediately suspending core alternation, positive reactivity changes, movement of irradiated fuel, and crane operation with loads over the fuel pool. With respect to suspending positive reactivity changes, operations that individually add limited, positive reactivity are acceptable when, combined with other actions that add negative reactivity, the overall net reactivity addition is zero or negative. For example, a positive reactivity addition caused by temperature fluctuations from inventory addition or temperature control fluctuations is acceptable if it is combined with a negative reactivity addition such that the overall, net reactivity addition is zero or negative. Refer to TS Bases 3/4.9.1, Boron Concentration, for limits on boron concentration and water temperature for MODE 6 action statements involving suspension of positive reactivity changes.

INSERT 3

3/4.8.2 DC SOURCES (continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std. 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.3 ONSITE POWER DISTRIBUTION

BACKGROUND

The onsite Class 1E AC, DC, and AC vital instrument bus electrical power distribution systems are divided by train into two redundant and independent power distribution subsystems.

The AC electrical power subsystem of each train consists of a Class 1E 4.16 kV emergency bus, 480-volt unit substations, and 120-volt vital instrument panels. Each 4.16 kV emergency bus has at least one separate and independent offsite source of power as well as a dedicated onsite diesel generator (DG) source. Each 4.16 kV emergency bus is normally energized from the unit auxiliary transformer (UAT). The opening of the UAT incoming line breaker, either manually or automatically, initiates an automatic transfer from the UAT to reserve auxiliary transformer (RAT), provided that the RAT transformer is energized. If all offsite sources are unavailable, the onsite

3/4.8.3 ONSITE POWER DISTRIBUTION (continued)

ACTIONS (continued)

MODES 5 and 6

With less than the minimum required on-site power distribution systems sources, the action statement requires immediately suspending core alterations, positive reactivity changes, or movement of irradiated fuel. With respect to suspending positive reactivity changes, operations that individually add limited, positive reactivity are acceptable when, combined with other actions that add negative reactivity, the overall net reactivity addition is zero or negative. For example, a positive reactivity addition caused by temperature fluctuations from inventory addition or temperature control fluctuations is acceptable if it is combined with a negative reactivity addition such that the overall, net reactivity addition is zero or negative. Refer to TS Bases 3/4.9.1, Boron Concentration, for limits on boron concentration and water temperature for MODE 6 action statements involving suspension of positive reactivity changes.

SURVEILLANCE REQUIREMENTS

Operability of the required electrical buses is confirmed by verifying correct breaker alignment and indicated voltage on the buses ~~at least once per seven days~~. ~~The seven-day frequency is based on the capability of the electrical systems and the indications available in the control room that alert the operator to electrical system malfunctions.~~

INSERT 4

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations are protected by deenergizing circuits not required during reactor operation. The OPERABILITY of the motor-operated valves thermal overload protection ensures that the thermal overload protection will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of the thermal overload protection are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

3/4.9.2 INSTRUMENTATION (Continued)

**SURVEILLANCE**

**SR 4.9.2.a**

SR 4.9.2.a is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency ~~of 12 hours~~ is consistent with ~~the CHANNEL CHECK Frequency~~ specified similarly ~~for~~ the same instruments ~~in~~ LCO 4.3.1.1.

INSERT 4

**SR4.9.2b**

SR 4.9.2.b is the performance of a CHANNEL CALIBRATION ~~every 18 months~~. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The CHANNEL CALIBRATION also includes verification of the audible count rate function. The 18 month Frequency is based on ~~the~~ need to perform this Surveillance under the conditions ~~that~~ apply during ~~a plant~~ outage. Operating ~~experience~~ has shown ~~these~~ components usually pass the Surveillance when performed at the 18 month ~~Frequency~~.

**REFERENCES**

1. 10 CFR 50, Appendix A. GDC 13, GDCP 26, GDC 28, and GDC 29.

2. FSAR, Section 15.4.6

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

Attachment 5

No Significant Hazards Consideration

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 Specification Task Force (TSTF) Traveler, TSTF-425, Revision 3 (Rev. 3) (ADAMS Accession No. ML09080642) 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 Specification 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. 071360456).

***Basis for proposed no significant hazards consideration***: As required by 10 CFR 50.91(a), the NextEra 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, NextEra 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, NextEra concludes that the requested change does not involve a significant hazards consideration as set forth in 10 CFR 50.92(c), Issuance of Amendment.

## Attachment 6

Cross-reference of TSTF 425 and Seabrook Surveillance Requirements

**Cross Reference Between TSTF-425 (NUREG-1431) Surveillance Requirements**
**And**
**Seabrook Surveillance Requirements**

(See footnotes on last page)

| Technical Specifications (TS) and Surveillance Requirements (SR) | TSTF 425 | Seabrook |
|---|---|---|
| **TS 3.1.1, Shutdown Margin** | | |
| Verify SDM in Modes 2 w/keff < 1; Modes 3, 4, and 5 | SR 3.1.1.1 | --- |
| *Verify SDM in Modes 1 and 2* | --- | 4.1.1.1.1.b |
| *Verify SDM in Modes 3 and 4* | --- | 4.1.1.1.1.e |
| *Verify SDM in Mode 5* | --- | 4.1.1.2.b |
| *Verify unborated sources isolated Modes 4, 5,and 6* | --- | 4.1.2.7 |
| **TS 3.1.2, Core Reactivity** | | |
| Verify reactivity $\underline{+}$1% | SR 3.1.2.1 | 4.1.1.1.2 |
| **TS 3.1.4, Rod Group Alignment Limits** | | |
| Verify rod position within alignment limit | SR 3.1.4.1 | 4.1.3.1.1 |
| Verify rod freedom of movement | SR 3.1.4.2 | 4.1.3.1.2 |
| *Verify rod drop time* | --- | 4.1.3.4.c |
| **TS 3.1.5, Shutdown Bank Insertion Limits** | | |
| Verify within insertion limits | SR 3.1.5.1 | 4.1.3.5.b |
| **TS 3.1.6, Control Bank Insertion limit** | | |
| Verify within insertion limits | SR 3.1.6.2 | 4.1.3.6 |
| Verify sequence and overlap limits met | SR 3.1.6.3 | --- |
| ***TS 3.1.3.1, Position Indication System*** | | |
| *Verify digital position indication operable by verifying digital and demand indications in agreement* | --- | 4.1.3.2 |
| *Verify digital position indication in agreement with demand indication when exercised* | --- | 4.1.3.3 |
| **TS 3.1.8, Physics Test Exceptions** | | |
| Verify RCS loop temperature | SR 3.1.8.2 | 4.10.3.3 |
| Verify thermal power $\leq$5% | SR 3.1.8.3 | 4.10.3.1 |
| *Verify thermal power < 85%* | --- | 4.10.2.1 |
| Verify SDM | SR 3.1.8.4 | 4.10.1.1 |
| *Perform Specs 4.2.2.2, 4.2.2.3, and 4.2.3.2* | --- | 4.10.2.2 |
| *Verify DRPI / Demand position agreement* | --- | 4.10.5 |
| **TS 3.2.1B, $F_Q(Z)$ Limits - RAOC** | | |
| Verify $F_Q(Z)$ limits - measured | SR 3.2.1.1 | 4.2.2.2.d.2 |
| Verify $F_Q^W(Z)$ limits | SR 3.2.1.2 | --- |

| Technical Specifications (TS) and Surveillance Requirements (SR) | TSTF 425 | Seabrook |
|---|---|---|
| **TS 3.2.2, $F^N_{\Delta H}$ Limits** | | |
| Verify $F^N_{\Delta H}$(Z) limits | SR 3.2.2.1 | 4.2.3.2 |
| **TS 3.2.3B, AFD Limits - RAOC** | | |
| Verify AFD within Limit | SR 3.2.3.1 | 4.2.1.1.a |
| **TS 3.2.4, QPTR** | | |
| Verify QPTR by calculation | SR 3.2.4.1 | 4.2.4.1.a |
| Verify QPTR w/ incore detectors | SR 3.2.4.2 | 4.2.4.2 |
| **TS 3.3.1, RTS Instrumentation** | | |
| Perform channel check | SR 3.3.1.1 | Table 4.3-1<br>Channel Check Column |
| Perform calorimetric – actual power adjust if > 2% | SR 3.3.1.2 | Table 4.3-1<br>Functional Unit (FU) 2.a |
| Compare and adjust NIS to incore ≥ 3% | SR 3.3.1.3 | Table 4.3-1, FU 2.a |
| Perform TADOT Rx trip breakers | SR 3.3.1.4 | Table 4.3-1 FUs, 19 & 21 |
| Perform Actuation Logic Test | SR 3.3.1.5 | Table 4.3-1 FU, Unit 20 |
| Calibrate NIS to incore | SR 3.3.1.6 | Table 4.3-1, FU 2.a |
| Perform COT - 184 days (Seabrook frequency 92 days) | SR 3.3.1.7 | Table 4.3-1<br>FU 2.a, 3, 7-13 |
| Perform COT - 184 days (Seabrook frequency 92 days) | SR 3.3.1.8 | Table 4.3-1 FU 6 |
| Perform TADOT | SR 3.3.1.9 | Table 4.3-1<br>FU 14, 15 |
| Perform channel calibration 18 months | SR 3.3.1.10 | Table 4.3-1<br>FU 9-16, 18.f |
| Perform channel calibration 18 months | SR 3.3.1.11 | Table 4.3-1, FU 2.a, 2.b, 3, 5, 6, 18.a, c, d, e |
| Perform channel calibration 18 months | SR 3.3.1.12 | Table 4.3-1, FU 7, 8 |
| Perform COT – 18 months | SR 3.3.1.13 | Table 4.3-1, FU 18.a, c-f |
| Perform TADOT – 18 months | SR 3.3.1.14 | Table 4.3-1, FU 1, 17 |
| *Perform 18 month channel calibration on P-7* | --- | Table 4.3-1, FU 18.b |
| *Perform 18 month COT on P-7* | --- | Table 4.3-1, FU 18.b |
| *Perform 18 month TADOT on RX trip bypass breaker* | --- | Table 4.3-1, FU 21 |
| Verify response time | SR 3.3.1.16 | 4.3.1.2 |
| **TS 3.3.2, ESFAS Instrumentation** | | |
| Perform channel check | SR 3.3.2.1 | Table 4.3-2<br>FU 1.c-e, 2.c, 3.b.3, 4.c-e, 5.b, 6.a, 7.c |
| *Perform channel check* | --- | Table 4.3-2<br>FU 3.c.4, 10.c |

| Technical Specifications (TS) and Surveillance Requirements (SR) | TSTF 425 | Seabrook |
|---|---|---|
| *Perform actuation logic test containment vent isolation* | --- | Table 4.3-2, FU 3.c.2 |
| *Perform actuation logic test P-14* | --- | Table 4.3-2, FU 10.c |
| Perform Actuation Logic Test | SR 3.3.2.2 | --- |
| *Perform Actuation Logic Test* | --- | Table 4.3-2<br>FU 10.b |
| Perform Actuation Logic Test | SR 3.3.2.3 | Table 4.3-2<br>FU 1.b, 2.b, 3.a.2, 3.b.2, 4.b, 5.a, 7.b, 8.a, |
| Perform Master Relay Test | SR 3.3.2.4 | Table 4.3-2<br>FU 1.b, 2.b, 3.a.2, 3.b.2, 4.b, 5.a, 7.b, 8.a, |
| *Perform Master Relay Test containment vent isolation* | --- | Table 4.3-2, FU 3.c.2 |
| *Perform Master Relay Test P-14* | --- | Table 4.3-2, FU 10.c |
| Perform COT | SR 3.3.2.5 | Table 4.3-2<br>FU 1.c-e, 2.c, 3.b.3, 4.c-e, 5.b, 6.a, 7.c, 8.b, 10.a |
| *Perform COT containment vent isolation* | --- | Table 4.3-2, FU 3.c.4 |
| *Perform COT P-14* | --- | Table 4.3-2, FU 10.c |
| Perform Slave Relay Test | SR 3.3.2.6 | Table 4.3-2<br>FU 1.b, 2.b, 3.a.2, 3.b.2, 4.b, 5.a, 7.b, 8.a, |
| *Perform slave relay test containment vent isolation* | --- | Table 4.3-2, FU 3.c.2 |
| *Perform slave relay test P-14* | --- | Table 4.3-2, FU 10.c |
| Perform TADOT | SR 3.3.2.7 | Table 4.3-2<br>FU 9a, 9.b |
| *Perform TADOT* | --- | Table 4.3-2<br>FU 8.b |
| Perform TADOT | SR 3.3.2.8 | Table 4.3-2<br>FU 1.a, 2.a, 3.a.1, 3.b.1, 4.a.1 |
| *Perform TADOT* | --- | Table 4.3-2<br>FU 3.c.1, 7.a.1, 7.a.2, |
| Perform channel calibration | SR 3.3.2.9 | Table 4.3-2<br>FU 1.c-e, 2.c, 3.b.3, 4.c-e, 5.b, 6.a, 7.c, 8.b, 9.a, 9.b, 10.a |
| *Perform channel calibration* | --- | Table 4.3-2<br>FU 3.c.4, 10.c |
| Verify response time | SR 3.3.2.10 | 4.3.2.2 |

| Technical Specifications (TS) and Surveillance Requirements (SR) | TSTF 425 | Seabrook |
|---|---|---|
| ***TS 3.3.3.1 Radiation Monitoring Instrumentation for Plant Operation*** | | |
| *Perform Channel Check, Channel Calibration, and Digital Channel Operational Test* | --- | 4.3.3.1 |
| **TS 3.3.3, PAM Instrumentation** | | |
| PAM channel check | SR 3.3.3.1 | 4.3.3.6.a |
| PAM channel calibration | SR 3.3.3.2 | 4.3.3.6.b |
| **TS 3.3.4, Remote Shutdown System** | | |
| Perform channel check | SR 3.3.4.1 | 4.3.3.5.1.a |
| Verify control and transfer switch function | SR 3.3.4.2 | 4.3.3.5.2 |
| Perform channel calibration | SR 3.3.4.3 | 4.3.3.5.1.b |
| Perform TADOT of reactor trip breaker | SR 3.3.4.4 | --- |
| ***TS 3.3.3.10, Explosive Gas Monitoring Instrumentation*** | | |
| *Channel Check, Channel Calibration, and Channel Operational Test* | --- | 4.3.3.10<br>Table 4.3-6 |
| **TS 3.3.5, LOP EDG Start Instrumentation** | | |
| Perform channel check | SR 3.3.5.1 | --- |
| Perform TADOT | SR 3.3.5.2 | Table 4.3-2, FU 9 |
| Perform channel calibration | SR 3.3.5.3 | Table 4.3-2, FU 9 |
| **TS 3.3.6, Containment Purge and Exhaust Isolation Instrumentation** | | |
| Perform channel check | SR 3.3.6.1 | Table 4.3-2, FU 3.c.4 |
| Perform Actuation Logic Test – 31 days | SR 3.3.6.2 | --- |
| Perform Master Relay Test – 3 days | SR 3.3.6.3 | --- |
| Perform actuation – 92 days | SR 3.3.6.4 | --- |
| Perform Master Relay Test -92 days | SR 3.3.6.5 | --- |
| Perform COT | SR 3.3.6.6 | Table 4.3-2, FU 3.c.4 |
| Perform Slave Relay Test | SR 3.3.6.7 | --- |
| Perform TADOT | SR 3.3.6.8 | --- |
| Perform channel calibration | SR 3.3.6.9 | Table 4.3-2, FU 3.c.4 |
| **TS 3.3.7, CREFS Actuation Instrumentation** | | |
| Perform channel check | SR 3.3.7.1 | 4.3.3.1, Table 4.3-3 FU 5 |
| Perform COT | SR 3.3.7.2 | 4.3.3.1, Table 4.3-3 FU 5 |
| Perform Actuation Logic Test – 31 days | SR 3.3.7.3 | --- |
| Perform Master Relay Test – 31 days | SR 3.3.7.4 | --- |
| Perform Actuation Logic Test – 92 days | SR 3.3.7.5 | --- |
| Perform Master Relay Test – 92 days | SR 3.3.7.6 | --- |
| Perform Slave Relay Test | SR 3.3.7.7 | --- |

| Technical Specifications (TS) and Surveillance Requirements (SR) | TSTF 425 | Seabrook |
|---|---|---|
| Perform TADOT | SR 3.3.7.8 | --- |
| Perform channel calibration | SR 3.3.7.9 | 4.3.3.1, Table 4.3-3 FU 5 |
| **TS 3.3.8, FBACS Actuation Instrumentation** | Note 1 | |
| **TS 3.3.9, BDPS** | Note 1 | |
| **TS 3.4.1, RCS Pressure, Temperature, and Flow DNB Limits** | | |
| Verify pressurizer pressure | SR 3.4.1.1 | 4.2.5.1 |
| Verify RCS average temperature | SR 3.4.1.2 | 4.2.5.1 |
| Verify RCS total flow | SR 3.4.1.3 | 4.2.5.1 |
| Verify RCS total flow w/ heat balance | SR 3.4.1.4 | --- |
| *Calibrate RCS total flow indicators* | --- | 4.2.5.2 |
| **TS 3.4.1, RCS Minimum Temperature for Criticality** | | |
| Verify RCS average temperature in each loop | SR 3.4.2.1 | --- |
| **TS 3.4.3, RCS P/T Limits** | | |
| Verify limits | SR 3.4.3.1 | 4.4.9.1 |
| **TS 3.4.4, RCS Loops - Modes 1 and 2** | | |
| Verify each loop operating | SR 3.4.4.1 | 4.4.1.1 |
| **TS 3.4.5, RCS Loops - Mode 3** | | |
| Verify required loops operating | SR 3.4.5.1 | 4.4.1.2.3 |
| Verify steam generator water level $\geq$ 17% | SR 3.4.5.2 | 4.4.1.2.2 |
| Verify breaker alignment and power available | SR 3.4.5.3 | 4.4.1.2.1 |
| **TS 3.4.6, RCS Loops - Mode 4** | | |
| Verify loop operation – RHR or RCS | SR 3.4.6.1 | 4.4.1.3.3 |
| Verify steam generator water Level $\geq$ 17% | SR 3.4.6.2 | 4.4.1.3.2 |
| Verify breaker alignment and power available | SR 3.4.6.3 | 4.4.1.3.1 |
| **TS 3.4.7, RCS Loops - Mode 5, Loops Filled** | | |
| Verify RHR loop operating | SR 3.4.7.1 | 4.4.1.4.1.2 |
| Verify steam generator water level $\geq$ 17% | SR 3.4.7.2 | 4.4.1.4.1.1 |
| Verify breaker alignment and power available RHR pumps | SR 3.4.7.3 | --- |
| **TS 3.4.8, RCS Loops - Mode 5, Loops Not Filled** | | |
| Verify RHR loop operating | SR 3.4.8.1 | 4.4.1.4.2 |
| Verify breaker alignment and power available RHR pumps | SR 3.4.8.2 | --- |
| **TS 3.4.9, Pressurizer** | | |
| Verify water Level | SR 3.4.9.1 | 4.4.3.1 |
| Verify heater capacity of required groups | SR 3.4.9.2 | 4.4.3.2 |
| Verify heater banks can be powered from emergency power supply | SR 3.4.9.3 | --- |

| Technical Specifications (TS) and Surveillance Requirements (SR) | TSTF 425 | Seabrook |
|---|---|---|
| **TS 3.4.11, Pressurizer PORVS** | | |
| Cycle each block valve | SR 3.4.11.1 | 4.4.4.2 |
| Cycle each PORV | SR 3.4.11.2 | 4.4.4.1.b |
| Cycle each SOV valve and check valve on the air accumulators in PORV control systems | SR 3.4.11.3 | --- |
| Verify PORVs and block valves can be powered from emergency power sources | SR 3.4.11.4 | --- |
| *Perform channel calibration* | --- | 4.4.4.1.a |
| **TS 3.4.12, LTOP System** | | |
| Verify only one HPI pump is capable of injecting into the RCS. | SR 3.4.12.1 | 4.5.3.2 |
| Verify a maximum of one charging pump is capable of injecting into the RCS. | SR 3.4.12.2 | 4.4.9.3.5 |
| Verify each accumulator is isolated. | SR 3.4.12.3 | 4.5.1.2 |
| Verify each RHR suction valve is open for each relief valve | SR 3.4.12.4 | 4.4.9.3.2.a<br>4.4.9.3.2.b |
| Verify required RCS vent [2.07] square inches open | SR 3.4.12.5 | 4.4.9.3.3 |
| Verify PORV block valve open for each required PORV. | SR 3.4.12.6 | 4.4.9.3.1.c |
| Verify RHR suction isolation valve is locked open with operator power removed for required RHR suction relief valve. | SR 3.4.12.7 | --- |
| Perform COT on each required PORV | SR 3.4.12.8 | 4.4.9.3.1.a |
| Perform channel calibration on each required PORV channel | SR 3.4.12.9 | 4.4.9.3.1.b |
| *Verify water level less than 36 inches below flange* | --- | 4.4.9.3.4 |
| **TS 3.4.13, RCS Operational Leakage** | | |
| Verify RCS operational leakage | SR 3.4.13.1 | 4.4.6.2.1.d |
| Verify SG leakage $\leq$150 gpd | SR 3.4.13.2 | 4.4.6.2.1.f |
| *Monitor reactor head flange leakoff system* | --- | 4.4.6.2.1.e |
| *Measure leakage to RCP seals* | --- | 4.4.6.2.1.c |
| **TS 3.4.14, RCS PIV Leakage** | | |
| Verify leakage from each is $\leq$ 0.5 gpm | SR 3.4.14.1 | 4.4.6.2.2.a |
| Verify RHR auto closure interlock prevents opening | SR 3.4.14.2 | 4.5.2.d.1 |
| Verify RHR auto closure interlock auto close | SR 3.4.14.3 | --- |
| **TS 3.4.15, RCS Leakage Detection Instrumentation** | | |
| Perform channel check – atmos rad monitor | SR 3.4.15.1 | 4.4.6.1.a.1 |
| Perform COT – atmos rad monitor | SR 3.4.15.2 | 4.4.6.1.a.2 |
| Perform channel calibration sump Monitor | SR 3.4.15.3 | 4.4.6.1.b |
| Perform channel calibration containment atmosphere | SR 3.4.15.4 | 4.4.6.1.a.3 |

| Technical Specifications (TS) and Surveillance Requirements (SR) | TSTF 425 | Seabrook |
|---|---|---|
| radioactivity monitor. | | |
| Perform channel calibration containment air cooler. | SR 3.4.15.5 | --- |
| **TS 3.4.16, RCS Specific Activity** | | |
| Verify RCS gross specific activity | SR 3.4.16.1 | 4.4.8, Table 4.4-3, item 1 |
| Verify reactor coolant Dose Equivalent 1-131 | SR 3.4.16.2 | 4.4.8, Table 4.4-3, item 2 |
| Determine E Bar | SR 3.4.16.3 | 4.4.8, Table 4.4-3, item 3 |
| **TS 3.4.17, RCS Loop isolation Valves** | Note 1 | |
| **TS 3.4.19, RCS Loops - Test Exceptions** | Note 1 | |
| ***TS 3.4.9.2, PZR Press Temp Limits*** | | |
| *Verify PZR temps and spray temp within limits* | --- | 4.4.9.2 |
| ***TS 3.4.11, RCS Vents*** | | |
| *Verify manual valves in flow path locked open* | --- | 4.4.11.2.a |
| *Cycle vent valve through one complete cycle* | --- | 4.4.11.2.b |
| *Verify flow through vent paths during venting* | --- | 4.4.11.2.c |
| **TS 3.5.1, Accumulators** | | |
| Verify accumulator isolation valve open | SR 3.5.1.1 | 4.5.1.1.a.2 |
| Verify borated water volume | SR 3.5.1.2 | 4.5.1.1.a.1 |
| Verify $N^2$ pressure | SR 3.5.1.3 | 4.5.1.1.a.1 |
| Verify boron concentration | SR 3.5.1.4 | 4.5.1.1.b.1 |
| Verify power removed from isolation valve | SR 3.5.1.5 | 4.5.1.1.c |
| *Verify isolation valve opens on RCS press signal* | --- | 4.5.1.1.d.1 |
| *Verify isolation valve opens in safety injection signal* | --- | 4.5.1.1.d.2 |
| ***TS 3.5.1.2, Accumulators - Shutdown*** | | |
| *Verify isolation valve closed with power removed* | --- | 4.5.1.2 |
| **TS 3.5.2, ECCS – Operating** | | |
| Verify valves aligned with power removed | SR 3.5.2.1 | 4.5.2.a |
| Verify valve position | SR 3.5.2.2 | 4.5.2.b.2 |
| Verify piping full of water | SR 3.5.2.3 | 4.5.2.b.1 |
| Verify automatic valve actuation | SR 3.5.2.5 | 4.5.2.e.1 |
| Verify automatic pump start | SR 3.5.2.6 | 4.5.2.e.2 |
| Verify throttle valve position | SR 3.5.2.7 | 4.5.2.g.2 |
| Inspection sump components | SR 3.5.2.8 | 4.5.2.d.2 |
| **TS 3.5.3.1, ECCS – Mode 4** | | |
| *Verify charging and SI pumps inoperable* | --- | 4.5.3.1.2 |
| **TS 3.5.4, RWST** | | |
| Verify water temperature | SR 3.5.4.1 | 4.5.4.b |
| Verify water volume | SR 3.5.4.2 | 4.5.4.a.1) |

| Technical Specifications (TS) and Surveillance Requirements (SR) | TSTF 425 | Seabrook |
|---|---|---|
| Verify boron concentration | SR 3.5.4.3 | 4.5.4.a.2) |
| **TS 3.5.5, Seal Injection Flow** | | |
| Verify throttle valve position | SR 3.5.5.1 | 4.4.6.2.1.c |
| **TS 3.5.6, Boron Injection Tank** | Note 1 | |
| **TS 3.6.2, Containment Air Locks** | | |
| Verify interlock operation | SR 3.6.2.2 | 4.6.1.3.b |
| **TS 3.6.3, Containment Isolation Valves** | | |
| Verify 42" purge valves sealed closed | SR 3.6.3.1 | --- |
| Verify 8" purge valves closed | SR 3.6.3.2 | 4.6.1.7.2 |
| Verify valves outside containment in correct position | SR 3.6.3.3 | 4.6.1.1.a |
| Verify isolation time of valves | SR 3.6.3.5 | --- |
| Cycle weight/spring loaded check valves | SR 3.6.3.6 | --- |
| Perform leak rate test of purge valves | SR 3.6.3.7 | --- |
| Verify automatic valves actuate to correct position | SR 3.6.3.8 | 4.6.3.2.a, b & c |
| Cycle non testable weight/spring loaded check valves | SR 3.6.3.9 | --- |
| Verify purge valves blocked | SR 3.6.3.10 | --- |
| **TS 3.6.4A, Containment Pressure** | | |
| Verify pressure | SR 3.6.4A.1 | 4.6.1.4 |
| **TS 3.6.5A, Containment Air Temperature** | | |
| Verify average air temperature | SR 3.6.5A.1 | 4.6.1.5 |
| **TS 3.6.6A, Containment Spray and Cooling Systems** | | |
| Verify valve position | SR 3.6.6A.1 | 4.6.2.1.a |
| Operate fan for $\geq$ 15 minutes | SR 3.6.6A.2 | --- |
| Verify cooling water flow | SR 3.6.6A.3 | --- |
| Verify automatic valve actuation | SR 3.6.6A.5 | 4.6.2.1.c.1 |
| Verify automatic pump start | SR 3.6.6A.6 | 4.6.2.1.c.2 |
| Verify auto start of cooling train | SR 3.6.6A.7 | --- |
| Verify each spray nozzle is unobstructed | SR 3.6.6A.8 | --- |
| **TS.3.6.7, Spray Additive System** | | |
| Verify valve position | SR 3.6.7.1 | 4.6.2.2.a |
| Verify tank volume | SR 3.6.7.2 | 4.6.2.2.b.1 |
| Verify tank solution concentration | SR 3.6.7.3 | 4.6.2.2.b.2 |
| Actuate each flow path valve | SR 3.6.7.4 | 4.6.2.2.c |
| Verify spray additive flow rate | SR 3.6.7.5 | --- |
| **TS 3.6.8, Shield Building** | Note 1 | |
| ***TS 3.6.5.2, Containment Enclosure Building Integrity*** | | |
| *Verify access opening doors are closed* | --- | 4.6.5.2 |

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| **TS 3.6.9, Hydrogen Mixing System** | | |
| Operate each train for ≥15 minutes | SR 3.6.9.1 | 4.6.4.3.a |
| Verify each train flow rate | SR 3.6.9.2 | 4.6.4.3.b |
| Verify start on actuation signal | SR 3.6.9.3 | --- |
| **TS 3.6.10, Hydrogen Ignition System** | Note 1 | |
| **TS.3.6.11, Iodine Cleanup System** | Note 1 | |
| **TS 3.6.13, Shield Building Air Cleanup System** | Note 1 | |
| ***TS 3.6.5.1, Containment Enclosure Emergency Air Cleanup System*** | | |
| *Operate system for ≥15 minutes* | --- | 4.6.5.1.a |
| *Cleanup unit penetration leakage testing* | --- | 4.6.5.1.b.1 |
| *Analysis of carbon sample* | --- | 4.6.5.1.b.2 |
| *Verify system flow rate* | --- | 4.6.5.1.b.3 |
| *Verify pressure drop* | --- | 4.6.5.1.d.1 |
| *Verify start on safety injection signal* | --- | 4.6.5.1.d.2 |
| *Verify filter cross-connect can be manually opened* | --- | 4.6.5.1.d.3 |
| *Verify system draws negative pressure in 4 minutes* | --- | 4.6.5.1.d.4 |
| **TS 3.6.14, Air Return System** | Note 1 | |
| **TS 3.6.15, Ice Bed** | Note 1 | |
| **TS 3.6.16, Ice Condenser Doors** | Note 1 | |
| **TS 3.6.17, Divider barrier Integrity** | Note 1 | |
| **TS 3.6.18, Containment Recirculation Drains** | Note 1 | |
| **TS 3.7.2, Main Steam Isolation Valves** | | |
| Actuate valves | SR 3.7.2.2 | --- |
| **TS 3.7.3, MFIVs and MFRVs** | Note 1 | |
| **TS 3.7.4, Atmospheric Dump Valves** | | |
| Cycle dump valves | SR 3.7.4.1 | --- |
| Cycle block valves | SR 3.7.4.2 | --- |
| *Verify nitrogen accumulator tank pressure* | --- | 4.7.1.6.a |
| **TS 3.7.5, AFW** | | |
| Verify valve position | SR 3.7.5.1 | 4.7.1.2.1.a.1 |
| *Verify auto valves fully open* | --- | 4.7.1.2.1.a.2 |
| *Verify valves operable for aligning startup feed pump* | --- | 4.7.1.2.1.a.3 |
| *Verify pump head* | --- | 4.7.1.2.1.b.1; 4.7.1.2.1.b.2; 4.7.1.2.1.b.3 |
| Verify auto valve actuation | SR 3.7.5.3 | 4.7.1.2.1.c.1 |
| Verify pump auto actuation | SR 3.7.5.4 | 4.7.1.2.1.c.2 |
| *Verify manual alignment within required time* | --- | 4.7.1.2.1.c.3 |

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| *Verify flow control valve closure on high flow* | --- | 4.7.1.2.1.c.4 |
| **TS 3.7.6, Condensate Storage Tank (CST)** | | |
| Verify volume of CST | SR 3.7.6.1 | 4.7.1.3.a |
| *Verify CST enclosure integrity* | --- | 4.7.1.3.b |
| **TS 3.7.7, Component Cooling Water System** | | |
| Verify valve position | SR 3.7.7.1 | 4.7.3.a |
| Verify valve actuation | SR 3.7.7.2 | 4.7.3.b. |
| Verify pump actuation | SR 3.7.7.3 | --- |
| **TS 3.7.8, Service Water System** | | |
| Verify valve position | SR 3.7.8.1 | 4.7.4.1.a; 4.7.4.2.a |
| Verify valve actuation | SR 3.7.8.2 | 4.7.4.1.b; 4.7.4.2.b.1; 4.7.4.2.b.2 |
| Verify pump actuation | SR 3.7.8.3 | 4.7.4.2.b.3 |
| ***TS 3.7.8, Sealed Source Contamination*** | | |
| *Test sealed sources* | --- | 4.7.8.2.a |
| **TS 3.7.9, Ultimate Heat Sink** | | |
| Verify water Level | SR 3.7.9.1 | 4.7.4.3; 4.7.4.4.a |
| Verify water Temperature | SR 3.7.9.2 | 4.7.4.4.b |
| Operate each cooling tower fan $\geq$15 minutes | SR 3.7.9.3 | 4.7.4.4.c.1 |
| Verify fan actuation | SR 3.7.9.4 | --- |
| *Verify portable pump in operational readiness state* | --- | 4.7.4.4.c.2 |
| *Verify portable pump develops flow $\geq$200 gpm* | --- | 4.7.4.4.d |
| **TS 3.7.10, Control Room Emergency Filtration System (CREFS)** | | |
| Operate each train for $\geq$10 hours | SR 3.7.10.1 | 4.7.6.1.a |
| Verify train actuation actual or simulated signal | SR 3.7.10.3 | 4.7.6.1.d.2.a; 4.7.6.1.d.2.b; 4.7.6.1.d.2.c; 4.7.6.1.d.3.a; 4.7.6.1.d.3.b; 4.7.6.1.d.3.c |
| Verify envelope pressurization | SR 3.7.10.4 | --- |
| *Cleanup unit penetration and bypass leakage testing* | --- | 4.7.6.1.b.1 |
| *Analysis of carbon sample* | --- | 4.7.6.1.b.2 |
| *Verify system flow rate* | --- | 4.7.6.1.b.3 |
| *Verify pressure drop* | --- | 4.7.6.1.d.1 |
| **TS 3.7.11, Control Room Emergency Air Temperature Control System (CREATCS)** | | |
| Verify train capacity | SR 3.7.11.1 | 4.7.6.2 |
| **TS 3.7.12, ECCS Pump Room Exhaust Air Cleanup System (PREACS)** Note 1 | | |
| **TS 3.7.1.3, Fuel Building Air Cleanup System (FBACS)** | | |
| Operate heaters | SR 3.7.13.1 | 4.9.12.a |
| Verify automatic train actuation | SR 3.7.13.3 | --- |
| Verify envelope negative pressure | SR 3.7.13.4 | 4.9.12.d.2 |

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| Verify bypass damper closure | SR 3.7.13.5 | --- |
| *Cleanup unit penetration leakage testing* | --- | 4.9.12.b.1 |
| *Analysis of carbon sample* | --- | 4.9.12.b.2 |
| *Verify system flow rate* | --- | 4.9.12.b.3 |
| *Verify pressure drop* | --- | 4.9.12.d.1 |
| *Verify cross connect valve can be manually opened* | --- | 4.9.12.d.3 |
| *Verify heaters dissipate 84 KW* | --- | 4.9.12.d.4 |
| **TS 3.7.14, Penetration Room Exhaust Air Cleanup System** | | Note 1 |
| **TS 3.7.15, Fuel Storage Pool Water Level** | | |
| Verify water Level | SR 3.7.15.1 | 4.9.11 |
| **TS 3.7.16, Fuel Storage Pool Boron** | | |
| Verify boron concentration | SR 3.7.16.1 | --- |
| **TS 3.7.18, Secondary Specific Activity** | | |
| Verify secondary activity | SR 3.7.18.1 | 4.7.1.4 |
| **TS 3.8.1, AC Sources –Operating** | | |
| Verify breaker alignment offsite circuits | SR 3.8.1.1 | 4.8.1.1.1.a |
| Verify EDG starts - achieves voltage & frequency | SR 3.8.1.2 | 4.8.1.1.2.a.5) |
| Synchronize and Load for > 60 minutes | SR 3.8.1.3 | 4.8.1.1.2.a.6) |
| Verify day tank Level | SR 3.8.1.4 | 4.8.1.1.2.a.1) |
| Remove accumulate water for day tank | SR 3.8.1.5 | 4.8.1.1.2.b |
| Verify operation of transfer pump | SR 3.8.1.6 | 4.8.1.1.2.a.3) |
| Verify EDG Starts – achieves voltage & frequency in 10 seconds | SR 3.8.1.7 | 4.8.1.1.2.e |
| Verify auto and manual transfer of AC power sources – offsite sources | SR 3.8.1.8 | 4.8.1.1.1.b |
| *Verify the EDG alignment for standby power* | --- | 4.8.1.1.2.a.7) |
| Verify largest load rejection | SR 3.8.1.9 | 4.8.1.1.2.f.2) |
| Verify EDG does not trip with load rejection | SR 3.8.1.10 | 4.8.1.1.2.f.3) |
| Verify de-energize, load Shed and re-energize emergency bus with loss of offsite power | SR 3.8.1.11 | 4.8.1.1.2.f.4)a) & b) |
| Verify EDG start on ESF signal | SR 3.8.1.12 | 4.8.1.1.2.f.5) |
| Verify EDG noncritical trips are bypassed | SR 3.8.1.13 | 4.8.1.1.2.f.6)c) |
| Run EDG for 24 hours | SR 3.8.1.14 | 4.8.1.1.2.f.7 |
| Verify EDG starts post operation – achieves voltage & frequency | SR 3.8.1.15 | 4.8.1.1.2.f.8 |
| Verify EDG synchronizes w/ offsite power and transfers load | SR 3.8.1.16 | 4.8.1.1.2.f.9) a), b) &c) |
| Verify ESF signal overrides test mode of EDG | SR 3.8.1.17 | 4.8.1.1.2.f.10) |
| Verify load sequencers are with design tolerance | SR 3.8.1.18 | 4.8.1.1.2.f.12) |

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| Verify EDG start on loss of offsite power with ESF | SR 3.8.1.19 | 4.8.1.1.2.f.6)a) & b) |
| Verify when started simultaneously from standby each EDGs reach rated voltage and frequency | SR 3.8.1.20 | 4.8.1.1.2.g |
| *Verify cooling tower pump starts on actuation signal with EDG loaded* | --- | 4.8.1.1.2.f.14 |
| *Manually connect startup feedwater pump with EDG loaded* | --- | 4.8.1.1.2.f.15 |
| **TS 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air** | | |
| Verify FO storage tank volume | SR 3.8.3.1 | 4.8.1.1.2.a.2) |
| Verify lube oil Inventory | SR 3.8.3.2 | 4.8.1.1.2.a.4) |
| Verify EDG air start receiver pressure | SR 3.8.3.4 | --- |
| Check and remove accumulate water from FO tanks | SR 3.8.3.5 | 4.8.1.1.2.c |
| *Verify operation of FO transfer pumps via the Installed cross-connect lines* | --- | 4.8.1.1.2.f.11 |
| **TS 3.8.4, DC Sources - Operating** | | |
| Verify battery terminal voltage | SR 3.8.4.1 | 4.8.2.1.a.2) |
| Verify station battery chargers capable of supplying [x]Amp for [y]Hours | SR 3.8.4.2 | 4.8.2.1.c.4) |
| Verify battery capacity | SR 3.8.4.3 | 4.8.2.1.d |
| *Verify No visible corrosion at terminal or connectors and resistance > xx ohms* | --- | 4.8.2.1.b.2) |
| *Verify no visual Indication of physical damage* | --- | 4.8.2.1.c.1) |
| *Cell-to-cell and terminal connections clean & tight* | --- | 4.8.2.1.c.2) |
| *Verify cell-to-cell Resistance is ≤ xx ohms* | --- | 4.8.2.1.c.3) |
| **TS 3.8.6, Battery Parameters** | | |
| Verify each battery float current is ≤ [2] amps. | SR 3.8.6.1 | 4.8.2.1.a.1 & b.1) |
| Verify each battery pilot cell voltage is ≥[2.07] V | SR 3.8.6.2 | 4.8.2.1.a.1 & b.1) |
| Verify each battery cell electrolyte level is ≥ to minimum design limits. | SR 3.8.6.3 | 4.8.2.1.a.1 & b.1) |
| Verify each battery pilot cell temperature ≥ to minimum design limits. | SR 3.8.6.4 | 4.8.2.1.b.3) |
| Verify each battery connected cell voltage is≥[2.07] V. | SR 3.8.6.5 | 4.8.2.1.b.1) |
| Verify station and EDG battery capacity - >80% after performance test | SR 3.8.6.6 | 4.8.2.1.e |
| **TS 3.8.7, Inverters - Operating** | | |
| Verify correct inverter voltage & alignment to required AC vital buses. | SR 3.8.7.1 | --- |
| **TS 3.8.8, Inverters - Shutdown** | | |
| Verify correct inverter voltage & alignment to required AC Vital buses. | SR 3.8.8.1 | --- |

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| **TS 3.8.9, Distribution System - Operating** | | |
| Verify correct breaker alignments and voltage to AC, DC, and AC vital bus electrical power distribution subsystems. | SR 3.8.9.1 | 4.8.3.1 |
| **TS 3.8.10, Distribution System - Shutdown** | | |
| Verify correct breaker alignments and voltage to AC, DC, and AC vital bus electrical power distribution subsystems. | SR 3.8.10.1 | 4.8.3.2 |
| ***TS 3.8.3.3, Trip Circuit for Inverter I-2A*** | | |
| *Verify trip circuit for inverter 2A operable* | --- | 4.8.3.3 |
| ***TS 3.8.4.1, AC Circuits Inside Primary Containment*** | | |
| *Verify containment load circuit breakers locked open* | --- | 4.8.4.1 |
| ***TS 3.8.4.3, Motor Operated Valves Thermal Overload Protection*** | | |
| *Verify thermal overloads operable* | --- | 4.8.4.3 |
| **TS 3.9.1, Boron Concentration** | | |
| Verify boron concentration is within the limit specified in COLR | SR 3.9.1.1 | 4.9.1.2 |
| **TS 3.9.2, Unborated Water Source Isolation Valves** | | |
| Verify each valve that Isolates unborated water sources is secured in the closed position | SR 3.9.2.1 | --- |
| **TS 3.9.3, Nuclear Instrumentation** | | |
| Perform channel check | SR 3.9.3.1 | 4.9.2.a |
| Perform channel calibration | SR 3.9.3.2 | 4.9.2.b |
| **TS 3.9.4, Containment Penetrations** | | |
| Verify each required containment penetration is in the required status. | SR 3.9.4.1 | 4.9.4.a |
| Verify each required containment purge and exhaust valve actuates to the isolation position on an actuated or simulated actuation signal. | SR 3.9.4.2 | 4.9.4.b |
| **TS 3.9.5, RHR and Coolant Circulation - *High Water Level*** | | |
| Verify one loop is in operation and circulating reactor coolant at a flow rate of > [2800] gpm. | SR 3.9.5.1 | 4.9.8.1 |
| **TS 3.9.6, RHR and Coolant Circulation - *Low Water Level*** | | |
| Verify one loop is in operation and circulating reactor coolant at a flow rate of > [2800] gpm. | SR 3.9.6.1 | 4.9.8.2 |
| Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation. | SR 3.9.6.2 | --- |
| **TS 3.9.7, Refueling Cavity Water Level** | | |
| Verify refueling cavity water level is ≥23 ft Above the Top of reactor vessel Flange. | SR 3.9.7.1 | 4.9.10 |

| | | |
|---|---|---|
| ***TS 3.11.1.4, Liquid Holdup Tanks*** | | |
| Determine quantity of radioactive material in outdoor tanks | --- | 4.11.1.4 |

(1) This table is provided for information only. The surveillance requirements are summarized and are not verbatim from the TS.
(2) Text in italics identifies Seabrook-specific surveillance requirements
(3) Note 1: This system, component, or TS is not in the Seabrook TS.