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U. S. Nuclear Regulatory Commission
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Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Facility Operating License Nos. DPR-53 and DPR-69

Subject: License Amendment Request: Adoption of Technical Specification Task Force Traveler (TSTF) – 425, Revision 3, Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b

Reference: (a) Technical Specification Task Force Traveler (TSTF) – 425, Revision 3, Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b

The Calvert Cliffs Nuclear Power Plant, LLC hereby requests an Amendment to Renewed Operating License Nos. DPR-53 and DPR-69 for Calvert Cliffs Unit Nos. 1 and 2, respectively, with the submittal of the proposed changes to the Technical Specifications to incorporate the Nuclear Regulatory Commission-approved Reference (a).

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

Attachment (1) provides a description and assessment of the proposed change, the requested confirmation of applicability and plant-specific verifications. Attachment (2) provides documentation of the probabilistic risk assessment technical adequacy. Attachment (3) provides the marked-up Technical Specification pages and Attachment (4) provides the marked-up Technical Specification Bases pages. A comparison of the Improved Standard Technical Specification Surveillance Frequencies and the Calvert Cliffs Technical Specification Surveillance Frequencies is presented in Attachment (5).

Calvert Cliffs requests approval of the proposed license amendment by April 30, 2015, with the amendment being implemented within 120 days. In accordance with 10 CFR 50.91(b)(1), a copy of this application is provided to the designated Maryland Official.

There are no regulatory commitments contained in this letter.

A001
NRR

Should you have questions regarding this matter, please contact Mr. Douglas E. Lauver at (410) 495-5219.

I declare under penalty of perjury that the foregoing is true and correct. Executed May 1, 2014.

Respectfully,

George H. Gellrich
Site Vice President

GHG/PSF/bjd

Attachments: (1) Description and Assessment of Proposed Changes
(2) PRA Technical Adequacy
(3) Marked-up Technical Specification Pages
(4) Marked-up Technical Specification Bases Pages
(5) Comparison Matrix

cc: NRC Project Manager, Calvert Cliffs
NRC Regional Administrator, Region I
NRC Resident Inspector, Calvert Cliffs
S. Gray, MD-DNR

ATTACHMENT (1)

DESCRIPTION AND ASSESSMENT OF PROPOSED CHANGES

1.0 DESCRIPTION

This letter is a request for an amendment to Renewed Operating Licenses DPR-53 and DPR-69 for Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2. The proposed amendment would modify the Technical Specifications by relocating specific surveillance frequencies to a licensee-controlled program with the implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies". Additionally, the change would add a new program, the Surveillance Frequency Control Program, to Technical Specification (TS) Section 5, Administrative Controls.

The changes are consistent with Nuclear Regulatory Commission (NRC) approved Technical Specification Task Force (TSTF)-425, Revision 3 (ADAMS Accession No. ML080280275). The Federal Register Notice published on July 6, 2009 (74 FR 31996) announced the availability of this TS improvement as part of the consolidated line item improvement process.

2.0 ASSESSMENT

2.1 Applicability of the Published Safety Evaluation

Calvert Cliffs has reviewed the NRC Safety Evaluation dated July 6, 2009 as part of the consolidated line item improvement process. This review included a review of the NRC staff's evaluation, TSTF-425, Revision 3, and the requirements specified in NEI 04-10, Revision 1 (ADAMS Accession No. ML071360456).

Attachment 2 includes CCNPP documentation with regard to probabilistic risk assessment technical adequacy consistent with the requirements of Regulatory Guide 1.200, Revision 1 (ADAMS Accession No. ML070240001), Section 4.2, and describes any probabilistic risk assessment models without NRC endorsed standards, including documentation of the quality characteristics of those models in accordance with Regulatory Guide 1.200.

Calvert Cliffs has concluded that the justifications presented in the TSTF proposal and the NRC staff's Safety Evaluation are applicable to CCNPP and justify this license amendment request for the incorporation of the changes to the Calvert Cliffs TSs.

2.2 Optional Changes and Variations

Calvert Cliffs is proposing variations or deviations as described below from the applicable Technical Specification changes described in TSTF-425, Revision 3 or the applicable portions of the NRC staff's model Safety Evaluation referenced in the Federal Register (74 FR 31996).

Note that Calvert Cliffs uses different numbering and titles than the Improved Standard Technical Specifications in several instances. Only TS changes consistent with Calvert Cliffs' design and TS are included. Attachment 5 provides specific information.

After NRC approval of TSTF-425, it was recognized that surveillance frequencies that have not been changed under the Surveillance Frequency Control Program (SFCP) may not be based on operating experience, equipment reliability or plant risk. Therefore, the TSTF and the NRC agreed that the TSTF-425 TS Bases insert, "The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program," should be revised to state, "The Surveillance Frequency is controlled under the Surveillance Frequency Control Program." The existing TS Bases information will be relocated to the licensee-controlled Surveillance Frequency Control Program.

The TSTF-425 TS Section 5.5.15 insert references NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies." The Calvert Cliffs TS 5.5.19 insert references NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies." This is an administrative deviation from TSTF-425 with no impact on the NRC's model safety evaluation dated July 6, 2009 (74 FR 31996).

The Comparison Matrix (Attachment 5) is provided for information and is a comparison between the NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," Surveillance Requirements included in TSTF-425 and the Calvert Cliffs SRs included in this license amendment request. The comparison includes a summary description of the referenced SR, which is provided for information purposes only and is not intended to be a verbatim description of the SR. The comparison matrix contains the following information:

- Calvert Cliffs SRs that have identical numbers to the corresponding NUREG-1432 SRs are not deviations from TSTF-425, with the exception of administrative deviations (if any) such as formatting and plant-specific Frequencies. These deviations are administrative with no impact on the NRC's model safety evaluation dated July 6, 2009 (74 FR 31996).
- Calvert Cliffs SRs that have different numbering than the NUREG-1432 SRs are an administrative deviation from TSTF-425 with no impact on the NRC's model safety evaluation dated July 6, 2009 (74 FR 31996).
- For NUREG-1432 SRs that are not contained in the Calvert Cliffs TS, the corresponding TSTF-425 mark-ups for the SRs are not applicable to Calvert Cliffs. This is an administrative deviation from TSTF-425 with no impact on the NRC's model safety evaluation dated July 6, 2009 (74 FR 31996).
- The following TS and associated SRs are Calvert Cliffs plant specific:
    - TS 3.4.17, Special Test Exception (STE) RCS Loops - Modes 4 and 5

  Calvert Cliffs has determined that these Calvert Cliffs plant specific SRs involve fixed periodic Frequencies and do not meet any of the four exclusion criteria of TSTF-425. In accordance with TSTF-425, changes to the Frequencies for SRs with periodic Frequencies that do not meet the exclusion criteria would be controlled under the SFCP. The SFCP provides the necessary administrative controls to require the SRs related to calibration, testing and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, the facility operation will be within safety limits, and that the limiting conditions for operation will be met. Change to frequencies in the SFCP would be evaluated using the methodology and probabilistic risk guidelines contained in NEI-04-10. The NEI 04-10 methodology includes qualitative considerations, risk analyses, sensitivity studies and bounding analyses, as necessary and recommended monitoring of the performance of systems, structures and components for which frequencies are changed to assure that the reduced testing does not adversely impact the SSCs. In addition, NEI 04-10 methodology satisfies the five key safety principles specified in Regulatory Guide 1.177 relative to changes in SR frequencies. Relocation of these frequencies is also consistent with TSTF-425 and with the NRC's model safety evaluation, including scope exclusions identified in Section 1.0 of the model safety evaluation.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Determination

Calvert Cliffs has reviewed the proposed no significant hazards determination published in the Federal Register on July 6, 2009 (74 FR 31996). Calvert Cliffs has concluded that the proposed no significant hazards consideration presented in the Federal Register notice is applicable to CCNPP, Unit Nos. 1 and 2 and is provided below, which satisfies the requirements of 10 CFR 50.90(a).

This change requests the adoption of an approved change to the standard technical specifications for Combustion Engineering plants (NUREG-1432) to allow relocation of specific Technical Specification (TS) surveillance frequencies to a licensee controlled program. The proposed change is described in Technical Specification Task Force (TSTF) traveler, TSTF-425, Revision 3 (ADAMS Accession No. ML080280275) 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 9, 2009 (74 FR 31996).

The proposed changes are consistent with Nuclear Regulatory Commission (NRC)-approved TSTF traveler, TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b." The proposed change relocates surveillance frequencies to a licensee controlled program, the Surveillance Frequency Control Program. This change is applicable to licensees using probabilistic risk guidelines contained in NRC-approved Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies," (ADAMS Accession No. ML071360456).

Calvert Cliffs has evaluated the proposed changes to the Technical Specifications using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration, in that operation of the facility in accordance with the proposed amendment would not:

i. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*

   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, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

ii. *Create the possibility of a new or different type of accident from any accident previously evaluated; or*

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, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

iii. *Involve a significant reduction in a margin of safety.*

Response: No.

The design, operation, testing methods, and acceptance criteria for systems, structures and components 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 updated final safety analysis report and the bases to the 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, Calvert Cliffs will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Revision 1 in accordance with the TS Surveillance Frequency Control Program. Nuclear Energy Institute 04-10, Revision 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, this proposed change does not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above, Calvert Cliffs concludes that the requested change involves no significant hazards consideration, as set forth in 10 CFR 50.92(c).

4.0 ENVIRONMENTAL CONSIDERATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

ATTACHMENT (2)

PRA TECHNICAL ADEQUACY

The Calvert Cliffs PRA Quality Statement in support of TSTF-425, Surveillance Frequency Control Program includes the following sections:

A. Internal Events PRA Quality
B. Fire PRA quality
C. Total CDF, LERF, and RG 1.174

A. Internal Events PRA Quality (27 Pages)

The PWROG performed a full scope internal events PRA peer review of CCNPP to determine compliance with ASME PRA Standard, RA-S-2008a and RG 1.200 (Reference 6.32) in June 2010. This review documented findings for all supporting requirements (SRs) which failed to meet at least Category II. The findings for that peer review are documented below in Table A-1. This table also includes the disposition, status, and impact on the PRA.

The peer review found that 97% of the SR's evaluated Met Capability Category II or better. There were 3 SR's that were noted as "not met" and 8 that were noted as Category 1. As noted in the peer review report the majority of the findings were documentation related. Of the 11 SR's which did not meet Category 2 or better, 7 were related to conservatisms or documentation in LERF and 2 were related to internal floods. There were 39 findings. Most of the findings have been addressed in the PRA model. No significant changes have been implemented in the internal events PRA. As there are no new methods applied, no follow on or focused peer reviews were required.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
1-16	AS-B3 SY-B6	Systems Analysis	Based on Sections 2.4 and 2.10 of the System Analysis Introduction Notebook (C0-SY-00, Rev. 0) this SR appears to be met. However, there is a potential issue related to this SR. Did not find reference to any engineering analysis needed to support Containment Air Cooler operation when this system is assumed to be available during LOSP when the containment heats up prior to electrical recovery. (This F&O originated from SR SY-B6)	Complete	The PRA Internal Events Accident Sequence Notebook, C0-AS-001, Section 3.3, has been updated with an engineering analysis of this issue. The analysis identifies that during the Loss of Offsite Power sequences, the Containment Air Coolers are credited for SBO conditions where the containment heats up, and then, after power recovery, the air coolers are credited for containment pressure and temperature control. For these accident sequences, offsite power is restored in one hour, and the containment pressure and corresponding saturation temperature remain well below containment design parameters that would challenge the CACs. Furthermore, failure of CACs is not risk significant, due to the potential availability of containment spray. REFERENCE C0-AS-001	No significant impact. Subsequent internal events accident sequence analysis shows that Containment Air Cooler operation is not challenged by containment heat up during LOOP accident sequences that credit the CACs for recovery.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
1-17	IFSO-A1 QU-E3	Internal Flooding	Examined Internal Flooding Notebook (C0-IF-001, Rev. 1) Sections 3.0 and 3.1. Part of the Internal Flood analysis may not be complete for assessing the Aux Feedwater Discharge Piping as a Flood Source. (This F&O originated from SR IFSO-A1)	Complete	An engineering analysis has subsequently been performed for AFW discharge piping flooding. The fraction of at-power time during which the AFW system is in operation 0.6% and the AFW Discharge Piping flood may be screened due to their low impact on CDF (<1E-9). REFERENCE C0-IF-001	No significant impact on CDF/LERF.
1-18	IFSO-A4 IFEV-A7	Internal Flooding	Examined Internal Flooding Notebook (C0-IF-001, Rev. 1) Section 3.3 and 5.3. Consideration of human-induced mechanisms as potential flood sources not clear. Regarding human-induced impacts on the flood frequency, Section 5.3 of the IF report states that they were included, but their inclusion should be better documented or referenced from IF (e.g., a sample calculation showing human contribution would be helpful). (This F&O originated from SR IFSO-A4)	Open	Human-induced impacts on the flood initiating event frequencies are not well documented. The issue has been captured in the PRA configuration control database (CRMP), but not yet addressed.	No impact. Maintenance-induced floods are included. Their contribution is based on industry data starting in 1985, and apportioned to various rooms based on the number of maintainable components in the room. This is documented in RAN 98-062, Rev 1, Section 5.0. The current flood analysis needs to be updated to reflect this.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
1-19	IFEV-B3 IFPP-B3 IFQU-B3 IFSN-B3 IFSO-B3	Internal Flooding	While some items are included in Section 7.0 of the IF report, many other instances of uncertainties and assumptions are cited throughout the report, but not included in the discussion of Section 7.0 nor are the implications of these other uncertainties and assumptions are discussed. (This F&O originated from SR IFPP-B3)	Open	In the Internal Flood notebook, the discussions on uncertainties and assumptions should be expanded. This issue has been captured in the PRA configuration control database (CRMP), but not yet addressed.	No significant impact as this is primarily a documentation issue.
1-25	DA-C7	Data	For the most part actual plant-specific data is used as a basis for the number of demands associated actual plant experiences (See basis for DA-C6), which includes both actual planned and unplanned activities. However, there are a few ESFAS testing and/or other logic channel testing that are not tracked via the plant computer. Created this F&O on non-documentation of ESFAS/logic train testing, which needs to include actual practice. (This F&O originated from SR DA-C7)	Complete	The ESFAS logic train testing has a very low risk significance and generally does not take the logic OOS. The train does go to 2-out-of-3 logic. Occurrences where the train is in 2-out-of-3 logic is incorporated into the PRA Data Analysis Notebook, C0-DA-001, Section 2.6 and 3.5. For the logic relays there is a RAW of <1.04 and Birnbaum on the order of 4E-07. Any logic relay unavailability that does not cause the ESFAS channel to be OOS and bypassed, is therefore of low significance. REFERENCE C0-DA-001	No significant impact on CDF/LERF.

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
2-7	IFPP-A5	Internal Flood	Section 2.3 provides a discussion that walkdowns used to confirm plant arrangement. The following note is contained in section 2.3: Unfortunately, the walk-down documentation from the original flooding analysis no longer exists. A plant walk-down was performed as a part of this analysis to provide familiarity with the plant design as well as confirm findings from the original walk-down. This walk-down is documented in a set of notes and photographs included in Appendix B. Walkdown photos for room 105A and 203 show equipment and potential flood propagation paths. However, there is not enough spatial information to develop specific targets for flood impingement or spray. (This F&O originated from SR IFPP-A5)	Complete	A walkdown was performed to assess the susceptibility to jet impingement or spray in rooms 105A and 203. All equipment is considered failed by spray or impingement for flood sources originating in the room. Notebook C0-IF-001 was updated with this additional documentation. REFERENCE C0-IF-001	No impact on CDF. This is an Internal Flood documentation issue.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
2-9	DA-D4	Data	Evidence of meeting this SR at CC-II/III is found in the PRA Data Notebook (C0-DA-001, Rev. 1) in Sections 2.1 and 2.7. Found inconsistencies in the value of total number components of different types (for both units) in Table 2-5 of the PRA Data Notebook with the actual total number for Calvert Cliffs. Also, found an inconsistency between the prior distribution and posterior distribution for SACM EDG fail to start in Table 2-6 of the Data Notebook. (This F&O originated from SR DA-D4)	Complete	Table 2-6 of the Data Notebook C0-DA-001 listed incorrect data and Bayesian update results for the SACMs. However, the correct values were used in the models for peer review. For the SACM EDGs in Table 2-6, the correct plant-specific data are in Table 2-5. Table 2-6 lists incorrect data and Bayesian update results for the SACMs. However, the correct values are used in the models. The above errors have been corrected in C0-DA-001. Other minor typographical errors were identified and corrected in the notebook. REFERENCE C0-DA-001	No impact on CDF/LERF. This was a documentation issue. The Internal Events model has been updated to include the correct data.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
3-3	SY-C2	Systems Analysis	Section 2.3 of each system notebook states that marked up plant system drawings are provided as supplements to the system notebook, which depicts the boundary of the system in terms of PRA modeling. The drawings are not in the notebooks. (This F&O originated from SR SY-C2)	Complete	Marked-up system boundary drawings were generated for each system notebook. Where Unit 1 and Unit 2 are similar, just the Unit 1 boundary is depicted. In addition, the system notebooks include drawing snippets, sketches, and descriptive text that also depict the system boundary. REFERENCE C0-SY-[All]	No impact on CDF/LERF. This is an Internal Events documentation issue that has been addressed.
3-5	SY-A11 SY-A6	Systems Analysis	The fault tree does not include potential failures of the AFW accumulator system. (This F&O originated from SR SY-A11)	Complete	A bounding sensitivity case was run to include failure of the AFW accumulators failing short-term AFW operation. This issue has an insignificant contribution to CDF. Short-term failure of the AFW operation is dominated by failure of electrical support systems and failure of active hardware (i.e., valves and instrumentation). The applicable system notebooks were updated. REFERENCES C0-SY-036 C0-SY-019 C0-SY-000	This finding does not significantly impact CDF/LERF. The random failure probability of the accumulators is two orders of magnitude lower than active hardware failures that support the same system function.

Table A-1 Internal Events PRA Peer Review – Facts and Observations						
F&O ID	**SR**	**Topic**	**Finding/Observation**	**Status**	**Disposition**	**Impact to CDF/LERF**
3-8	SY-C1 SY-A13	Systems Analysis	Several system notebooks were reviewed (AFW, EDG, SI, 120 VAC electrical, etc.). In general, the documentation is complete and thorough. In most cases it clearly follows the RG 1.200 SRs. In some places, assumptions were imbedded in the documentation without sufficient reference or justification. Examples include: SI notebook page 11, last bullet 'Only one of the three HPSI pumps functions - For a cold leg break, it is assumed that only one-fourth pump discharge is spilled via the break. For a hot leg break, the entire pump discharge reaches the core.' SI notebook page 12, 2nd bullet 'The maximum time assumed for operation for the safety injection pumps is 30 seconds following SIAS initiation.' C0-SY-000 states that each system notebook addresses flow diversions (where applicable) in section 3.4.d. Although flow diversions appear to be addressed (for example, the SW notebook talks about flow diversion), there is no consistent discussion in each system notebook. (This F&O originated from SR SY-C1)	Complete	Some new flow diversions were identified as part of the Fire PRA Multiple Spurious Operation review, and these were added to the system models and system notebooks. Furthermore, a comprehensive review of PRA mechanical systems notebooks and drawings was performed to identify and document potential flow diversions. Flow diversion discussions were added to Sections 3.4.d of the applicable system notebooks. The success criteria for safety injection pumps is developed, justified and documented in the Success Criteria Notebook, C0-SC-001.	Flow diversion has been documented and are in the internal events PRA model. The safety injection success criteria was verified to be documented in the Success Criteria notebook. These were documentation issues and do not affect CDF/LERF.

Table A-1 Internal Events PRA Peer Review – Facts and Observations						
F&O ID	**SR**	**Topic**	**Finding/Observation**	**Status**	**Disposition**	**Impact to CDF/LERF**
3-9	DA-B1	Data	DA notebook table 2-5 contains the grouping of components for plant specific failure data. Many of the groupings appear to take into account differences in such things as size, type, mission type (e.g., FW TDP run vs. AFW TDP standby). However, in some cases, it is not clear what the basis for the grouping is. For example, SW MDP RUN and SRW MDP RUN are grouped together even though they are of different service conditions (salt water vs. clean water), voltages (480 VAC vs. 4160 VAC), size, etc. Similarly, AFW MDP is included with HPSI MDP and LPSI MDP, even though the two SI pumps are pumping borated water, while the AFW pump is pumping condensate grade water. No documentation of the appropriateness of these groupings is provided. (This F&O originated from SR DA-B1)	Complete	The model has been updated to add additional component types and failure modes to better reflect service conditions. Service Water and Salt Water pumps were broken out. AFW pumps and Safety Injection pumps were broken out. This resulted in changes to the associated failure rates. The change has been reflected in the Data Notebook, C0-DA-001. REFERENCE C0-DA-001	The internal events model includes the updated data and failure modes.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
3-11	QU-B7	Quantification	The mutually exclusive cutsets for each system are described in the system notebook section 3.4.e. Several SY notebooks were reviewed to determine appropriateness of the mutually exclusive cutsets. All appeared reasonable. A review was performed of the MUTEX gate within the fault tree model and the appropriate combinations identified in the SY notebooks appear have been included in the model. There are two gates under the MUTEX gate which contain mutually exclusive cutsets which are not documented in the system notebooks. While the majority of these are intuitively obvious (e.g., 11 Steam Generator Tube Rupture occurs as an IE AND 12 Steam Generator Tube Rupture occurs as an IE), these should be included in an appropriate system notebook. (This F&O originated from SR QU-B7)	Complete	A comprehensive review of mutual exclusive modeling was performed. Each system notebook and each system model was reviewed to validate the appropriateness of the modeling and reconcile any differences, and to verify that a documented basis exists for each mutually exclusive event. The PRA model was updated to reflect new, deleted, or re-organized mutually exclusive modeling identified as part of this review. REFERENCE SY-C0-[ALL]	No impact on CDF/LERF. The PRA model has been updated where needed.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
3-12	QU-D3	Quantification	A review of the top cutsets from each event tree was performed. The utility stated that during this review, cutsets were reviewed to determine if any mutually exclusive events were contained within cutsets, if any flag settings were inappropriate or if any recoveries were overlooked or added inappropriately. A review of a sampling of cutsets did not indicate any inappropriate results. However, the QU notebook does not include a discussion of this review. (This F&O originated from SR QU-D3)	Complete	Documentation of the cutset reviews was presented to the peer review team; although, the documentation was separate from the formal QU notebook package. A note was added to the QU notebook directing the reader to the location of the cutset review notes and spreadsheets. The PRA configuration control procedure, CNG-CM-1.01-3003, requires a review of cutsets for PRA changes. In practice, the top CDF and LERF cutsets are examined for even the most innocuous model changes. REFERENCES CNG-CM-1.01-3003 C0-QU-001 C0-FRQ-001	No impact on CDF/LERF. The original internal events cutset review notes have now been archived.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
4-5	IE-A10 SY-A10 IE-C3 SC-A4	Initiating Events	The only mention in C0-SC-001 of shared systems between the units is the SBO EDG, noted in Section 4.1.2. It states that the SBO diesel can power any one bus on either unit. However, in the CAFTA model, there is an assumed bus preference of 11, then 24, then 12, then 23.* This is noted in the EDG system notebook but no basis is provided. The procedures do not actually have a preference, which yields a potentially non-conservative analysis. For example, if there is a LOOP, the U2 diesels fail to start and the U1 diesels fail to run after 1 hour. The SBO diesel would then be aligned to U2, and it is non-conservative to give the U1 bus 11 full credit. If such non-conservatism is negligible, some analysis should be performed to demonstrate this. (This F&O originated from SR IE-A10) *Note: Peer review finding was not precise. It should have stated bus preference for Unit 1 is 11, then 24, and for Unit 2, is 24 then 21.	Complete	To address this finding, the Diesel Generator modeling was updated as described in Appendix H of C0-SY-023-024, PRA DG System Notebook. EOP-7 directs to align the 0C DG to the unit with redundant safety equipment out-of-service, with a goal to restore at least one 4KV bus. Since 4KV Buses 11 and 24 support AFW, those busses would have a preference over Busses 14 and 21, all else being equal. No unit preference is modeled. If there is a conflict in the order-of-preference, for example, both 4KV Bus 11 and 4KV Bus 24 are not powered, then a 50-50 probability is assumed as to the preferred bus. REFERENCE C0-SY-023-024	No impact. The finding has been addressed in the Internal Events model.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
4-12	HR-C1	Human Reliability	One basic event calculated in the appendix (ESF0HFCISZDEFG) was not included in the fault tree models. CCNPP staff noted that it had previously been modeled, but inadvertently deleted in an update. (This F&O originated from SR HR-C1)	Complete	The basic event has been added to the model. A sensitivity run with the basic event included the current model showed no increase in risk. The system notebook C0-SY-048 was updated. REFERENCE C0-SY-048	No impact. The missing basic event has been added to the internal events model.
4-15	IFEV-A6	Internal Flooding	The internal flooding analysis did not have a formal process to gather plant specific design information, operating practices, etc. that could potentially affect the generic flooding frequencies. In response to an NRC RAI on the CCNPP ISI program plan, CCNPP mentioned a review of Condition Reports that did not find any items that would increase the flooding frequency. The CR review meets part of the requirement, but the SR also calls for reviews of plant design, operating practices, etc. that should be considered. The evaluation should be documented in the PRA. (This F&O originated from SR IFEV-A6)	Open	CCNPP is typical of a PWR that uses raw water for cooling. Raw water, in our case salt water from the Chesapeake Bay, is used to cool the Component Cooling Heat exchangers, Service Water Heat exchangers, main condensers, and ECCS Pump Room air coolers. Salt water floods comprise the greatest flood risk as there is an infinite volume. However, these systems are designed, as typical, to be able to isolate a heat exchanger or condenser for draining and cleaning. CCNPP is unlikely to have any plant specific design or operating practices that would affect the generic flooding frequencies.	The review of condition reports did not identify any design issues or operating practices that would affect the generic flooding frequencies. Therefore, no significant impact is expected.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
4-19	LE-C13 LE-F3 LE-G4	Large Early Release	The sources of uncertainty are well identified in Table 5-1 of the LE notebook and quantified in Table 5-2 of the QU notebook. However, no discussion of the uncertainties or insights from them is provided. For example, Sensitivity 1 shows a 74% reduction in LERF, but this large reduction is not investigated. Also, conservatisms in the ISLOCA analyses were discussed in the AS review. SGTR was treated in an overly conservative manner by categorizing all SGTR as LERF. (This F&O originated from SR LE-F3)	Complete	Dominant LERF cutsets were reviewed to identify uncertainties that could be addressed. Two changes have been implemented to address significant uncertainties and reduced LERF. First, a reverse-flow check valve in the CVCS Letdown line was credited as a potential ISLOCA recovery. Second, a new human action was added with realistic timing for Steam Generator isolation and RCS depressurization on a SGTR. These and less significant model updates resulted in a LERF-to-CDF ratio change from approximately 17% to approximately 10%. This newer ratio is in the typical range for other PWRs. REFERENCE C0-LE-001	No significant impact. The dominant LERF contributors were reviewed and model changes implemented. The Calvert Cliffs LERF contribution is now similar to other PWRs.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
4-20	LE-F1 LE-G3	Large Early Release	The relative contribution to LERF is presented in the QU notebook by PDS and by initiating event, but not by accident progression sequence, phenomena, containment challenges or containment failure mode. (This F&O originated from SR LE-G3)	Complete	The contributions to LERF are documented in the Quantification Notebook and are noted as such in the Level 2 Notebook. Accident progression sequences are located in Section 4.2.3 and Appendix C. The Level 2 notebook has been updated to point to additional phenomena and containment challenges and failure mode Tables/Figures in the QU Notebook. REFERENCES C0-QU-001 C0-LE-001	No impact. This is an internal events documentation issue.
4-21	LE-G5	Large Early Release	The LE notebook states that limitations in the LE analysis that could impact applications are documented in the QU notebook, but it is not. Given the conservative modeling of SGTR and ISLOCA, the impact on applications should include assessment of how this conservatism can skew the LERF results. (This F&O originated from SR LE-G5)	Complete	Section 5.5.2.7 of C0-LE-001, Revision 2 - added discussion of this issue and how it was addressed. Model changes included crediting the CVCS reverse flow check valve and adding a new human action with realistic timing for isolating the steam generators and de-pressurizing the RCS on a SGTR. REFERENCE C0-LE-001	No significant impact.

Table A-1 Internal Events PRA Peer Review – Facts and Observations						
F&O ID	**SR**	**Topic**	**Finding/Observation**	**Status**	**Disposition**	**Impact to CDF/LERF**
4-22	LE-C10 LE-C12 LE-F2 LE-C3	Large Early Release	The LERF contributors have not been reviewed for reasonableness (per SR LE-F2). The QU notebook discusses the top 20 LERF cutsets (which total 73% of the total LERF). It notes conservatism in the cutsets and says it will be evaluated in Section 5.2, but is not. Section 4.3.6 of the QU notebook compares the total LERF of CCNPP to St. Lucie, but does not even break the results down by contributor (e.g., SGTR, ISLOCA, etc.). Also, the ASME PRA Standard SRs C-3, C-10 and C-13 require a review of the LERF results for conservatism in the following areas: 1. Engineering analyses to support continued equipment operation or operator actions during severe accident progression that could reduce the LERF. 2. Engineering analyses to support continued equipment operation or operations after containment failure. 3. Potential credit for repair of equipment. No such review has been performed, despite the large conservatism noted in the containment bypasses. (This F&O originated from SR LE-F2)	Complete	The LERF results were reviewed for conservatisms as described in the SRs. After conservatisms were addressed (see discussion for F&0 4-19 above), no significant issues were identified. REFERENCE C0-LE-001	No significant impact. The dominant LERF contributors were reviewed and model changes implemented. The Calvert Cliffs LERF contribution is now similar to other PWRs.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
5-10	LE-D7	Large Early Release	Following the failure of one or more containment penetrations to isolate on CIAS, a feasible operator action is to manually close the failed valves from the Main Control Room. (This F&O originated from SR LE-D7)	Complete	The merits have been considered of adding an operator action in order close containment penetration from the Main Control Room to recover from a containment isolation failure. A review of cutsets shows that a recovery is not feasible for top LERF sequences, because the sequence includes either 1) a loss of CR indication, 2) includes a station black-out condition, or 3) includes non-recoverable pipe breaks. REFERENCES C0-LE-001 Attachment S	Modeling of an operator action to manually close failed valves from the main control room would not significantly reduce LERF, as such an action is not feasible for the significant sequences where containment isolation has failed.
5-17	IE-C1 IE-C13 IE-C4	Initiating Events	Bayesian updates of non-time-based LOCA data were improper. The small and medium LOCA frequencies were obtained from draft NUREG 1829 then Bayesian updated (in App E) with CCNPP experience from 2004 to 2008. The Very Small LOCA prior having alpha = 0.4, Mean = 1.57E-03; was Bayesian updated to a Posterior having a mean value of 7.02E-04. This represents an excessive drop associated with CCNPP experience of 4 to 5 years. Similarly, the Small and Medium LOCAs were Bayesian updated with the whole industry	Complete	CENG understands the general concern on Bayesian updating of rare events. However, the method used was based on a white paper developed by industry experts regarding LOCA frequencies. These experts included INL, NRC and Industry experts. In addition, the approach used for the Calvert PRA was the same as used for the NRC SPAR model. This issue is captured in the PRA configuration control database	No impact. The approach used for LOCA frequencies has been validated by industry experts and is the same approach as was used for the NRC's SPAR model.

Table A-1 Internal Events PRA Peer Review – Facts and Observations						
F&O ID	**SR**	**Topic**	**Finding/Observation**	**Status**	**Disposition**	**Impact to CDF/LERF**
			experience rcy data. The draft NUREG 1829 LOCA frequencies were obtained from expert elicitations (not time-based) that included crack propagation analysis. The Bayesian update for VSLOCA used the Alpha parameter and the mean value to justify that the prior mean was based on 255 rcy. This may not have been the basis for the expert elicitations in NUREG 1829. Also, the Medium LOCA frequency may be classified as extremely rare event. It would require no Bayesian updating. The current CCNPP SLOCA and MLOCA frequencies are very close even though the source data in NUREG 1829 indicates a negative exponential drop in these frequencies. (This F&O originated from SR IE-C1) (Note: rcy – reactor year)		(CRMP). REFERENCE C0-IE-001	
5-18	IE-C2 IE-C7	Initiating Events	Justify the exclusion of LOOP event at CCNPP in 1987. No time trend analysis was provided to justify the exclusion. (This F&O originated from SR IE-C2)	Complete	The event is not counted following guidance provided in NUREG/CR-6928, based upon trend analysis. A full discussion is included in the Initiating Event notebook, C0-IE-001. REFERENCE C0-IE-001	No impact. The data analysis is acceptable.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
5-23	HR-A2	Human Reliability	The Pre-Initiator HRAs did not include the miscalibration of SIT pressure. For example, in the event where SIT pressure is miscalibrated high, various accident scenarios requiring SI are negatively impacted. Add SIT pressure miscalibrated high or, justify no impact on CDF / LERF. (This F&O originated from SR HR-A2)	Complete	It is agreed that the miscalibration of SIT pressure could have a negative impact on various accident scenarios involving LLOCA and VLLOCA initiators. However, this instrumentation is not modeled explicitly and is therefore deemed included within the component boundary for the SIT. As such the miscalibration probability would be included in the SIT unavailability. REFERENCE C0-HR-001	No impact. Given the pressure of the CCNPP SITs they are only required and provide significant benefit on Large LOCAs. The frequency of a Large LOCA times the pre-initiator frequency is negligible.
5-25	SC-C1 HR-I2 SC-C2	Success Criteria	Simplify the traceability of Tsw. In the post initiator HRA details, the HRA success criteria are often provided as a positive re-statement of the HRA title. And, the consequence of failure is often stated as core damage. Consider adding Tsw to the success criteria and linking that to the PCTran case where Tsw was developed. Also, in the SC report (Table B-3), consider adding the actual time to core uncovery (or core damage) instead of providing a "Yes" entry in the column of "core damage?" (This F&O originated from SR HR-I2)	Complete	Where applicable, the Tsw of each HFE that could be traced to the Success Criteria notebook (CS-SC-001) was updated and referenced in the HRA Calculator. C0-HR-001 was also updated. REFERENCE C0-HR-001	No impact. This is an internal events documentation finding.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
5-30	LE-D1 LE-B2	Large Early Release	Section 3.2.11 discussed the containment challenge from Hydrogen Combustion. It concluded that the challenge may be significant for some accident scenarios. The CCNPP entry in Table 6.11-2 of the Level 2 WCAP showed a potentially significant impact from Hydrogen burn. Provide an estimate of the impact of Hydrogen burn on containment pressure. Use an accident scenario that is likely to produce larger amounts of H2 with failed containment spray. The optimal time to estimate the impact of Hydrogen burn is approximately at 2 hours which is the time when the EOF and TSC personnel have convened and are ready to guide the Main Control Room into periodic Hydrogen burns before the formation of explosive mixtures. (This F&O originated from SR LE-D1)	Complete	CCNPP's Level 2 PRA follows the analysis in WCAP-16341-P, Simplified Level 2 Modeling Guidelines. In the industry-supported analysis, the percentage of cladding oxidation is the main factor used to develop a maximum H_2 concentration in the containment, and, in turn, a containment pressure is calculated if the H_2 completely burns. These are then mapped to site-specific containment failure probabilities. A simplifying assumption is made that "no pre-burning of hydrogen generated in the core melt progression is considered." Calvert Cliffs' severe action management procedures do include actions to reduce H_2 concentration in the containment, but these actions are not credited in the PRA model. Also, Containment Spray is not questioned for the LERF accident sequences. Containment Spray is a factor in LATE containment failure accident sequences.	No impact. The methodology in WCAP-16341-P is appropriate for Calvert Cliffs level 2 analysis.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
					REFERENCE C0-LE-001	
5-31	DA-D4	Data	The summary table for Bayesian updated parameters (on Page 53 of the PRA Data Notebook, C0-DA-001, Rev. 1) shows the CS-MDP was Bayesian updated with plant experience containing 1 failure and Zero run-hours. The CCNPP PRA staff responded to this issue as an isolated case. There is an actual FTR > 1 hr (This F&O originated from SR DA-D4)	Complete	The aforementioned footnote was incorporated into Table 2-6 of C0-DA-001. REFERENCE C0-DA-001	No impact. This is a minor internal events documentation issue and no changes were required for the CS-MDP failure rate.

Table A-1 Internal Events PRA Peer Review – Facts and Observations						
F&O ID	**SR**	**Topic**	**Finding/Observation**	**Status**	**Disposition**	**Impact to CDF/LERF**
6-3	SC-B2	Success Criteria	Expert judgment was not used as the sole basis for any success criteria. However, upon inspection of the PCTran run tables in the SC report appendices, many instances of surrogate or inferred results were found. Instead of running specific PCTran calculations to cover the whole SLOCA break size spectrum, intermediate break sizes have been calculated supplemented with expert judgment to derive limiting time delay for operators to actuate SI (30 min) or limiting time delay for OTCC (SGL<350'+10min). (This F&O originated from SR SC-B2)	Complete	The approach for SLOCA break size analysis is discussed in the Success Criteria notebook. Furthermore, a review was conducted of this issue; in addition, TH analyses were completed to verify the break-size ranges. It was found that the computer simulations adequately represented the various break-size ranges. Notwithstanding the argument above, two additional SLOCA computer simulation runs were made and documented in Success Criteria Notebook C0-SC-002 – PCTran simulations. Success Criteria Notebook C0-SC-001 was updated to clarify and incorporate the additional reviews. REFERENCE C0-SC-001	No impact.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
6-5	SY-A20	Systems Analysis	When appropriate, the simultaneous unavailability within a system is documented in the system notebooks and included in the PRA model. However, a further review of these items is required for completeness. (This F&O originated from SR SY-A20)	Complete	AFW basic event AFW0TMMAINT6-F7 was determined to not be needed in the plant model. The basic event was removed. All remaining AFW equipment unavailability events in the model and notebooks were reviewed for consistency. AFW0TMMAINT-TF was determined to be modeled correctly, its description was found to be in error in the system notebook. Notebook C0-SY-036 was updated. A review for concurrent maintenance was previously performed and documented in the Data Notebook. REFERENCE C0-SY-036	No impact. The offending basic event was removed from the model. A review did not discover other missing or incorrect simultaneous unavailability events.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
6-8	HR-H2	Human Reliability	Some recovery actions included in the model (thus credited) are set to screening values. In the HEP evaluation (appendices of the HR report) there are no indications that procedures, training, or other shaping factors are available on a plant-specific basis. (This F&O originated from SR HR-H2)	Complete	For each screening HRA, the internal events analysis was updated to include a specific reference to the earlier HRA analysis. Included are the applicable success criteria for each recovery. Refer to C0-HR-001, Internal Events Human Reliability Analysis, and the associated HRA Calculator file. REFERENCES C0-HR-001 C0-HRA-001	No impact. The documentation for internal events HRAs was updated to address this finding.
6-9	HR-I1	Human Reliability	The HR report is well documented in general and will facilitate upgrades, however, some basic event names are not consistent between the HR report and the system notebooks. (This F&O originated from SR HR-I1)	Complete	Updated the notebooks in the reference section so HRA designator names and descriptions are the same in the HR Calculator, HR notebook, CAFTA Model 6.0. Changes included adding the "-B" extension and removing the "(-2)" event where applicable. REFERENCES C0-HR-001 C0-SY-[Many]	No impact. This is a documentation finding. HRA names in the model and notebook are now consistent.

Table A-1 Internal Events PRA Peer Review – Facts and Observations						
F&O ID	**SR**	**Topic**	**Finding/Observation**	**Status**	**Disposition**	**Impact to CDF/LERF**
6-10	IFPP-A2 IFSN-A2	Internal Flooding	Plant design features such as open rooms or as built divisions are used to define the flood areas and was well documented. More detail is needed as to why the containment buildings were screened from the analysis. (This F&O originated from SR IFPP-A2)	Complete	The Internal Flood notebook has been updated to incorporate an analysis describing the screening of the containment building from flooding analysis. Essentially, the containment is designed for LOCA condition, which screens reactor coolant system and related piping system. Other piping systems have limited inventory, are normally isolated, or have a low flow rate. Reference C0-IF-001. REFERENCE C0-IF-001	No impact.
6-14	IFSO-B1 IFSN-A9	Internal Flooding	While the flooding calculations have been performed and are thought to be correct and well done, additional documentation of data would enhance the IF report. It appears that the input reports and references are based on poorly documented or non-officially revisioned reports and information sources. (This F&O originated from SR IFSN-A9)	Open	This is a documentation finding for the internal floods notebook. The issue has been captured in the PRA configuration control database (CRMP), but not yet closed-out.	No impact as this is a documentation issue.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
6-16	IFQU-A11 IFPP-B2	Internal Flooding	Walkdowns have been conducted and are documented in Appendix B of the IF report. It is stated in the IF report that prior information is no longer available; this fact should be corrected as required for analysis updates and information verifications. (This F&O originated from SR IFQU-A11)	Open	This is an internal floods documentation finding. The finding has been captured in the PRA configuration control database (CRMP), but not yet addressed.	No impact. This is a documentation issue.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
6-17	IFQU-A10	Internal Flooding	By including the flooding events under the transient fault tree, the LERF impacts are automatically accounted for in the same manner as the general transient events in the LERF analysis. Very little documentation is found related to the IF analysis in the LE report, although the IF report states that the LERF impacts due to flooding are documented and analyzed in the LE report. (This F&O originated from SR IFQU-A10)	Open	This internal flood issue is captured in the PRA configuration control database (CRMP), but not yet addressed. There is some potential that the analysis, in addition to providing the required documentation, may drive changes to the model.	This finding may drive some changes to the model for internal floods but it is not expected to be significant. The level of modeling detail in the CCPRA is sufficiently robust such that the model logic for flood impacts propagate appropriately through the system fault trees so that the equivalent general transient initiator (e.g. loss of CCW) is appropriately defined in the transient fault tree. In addition, cutset reviews have not revealed the current modeling to be deficient in this regard.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
6-18	HR-H2	Human Reliability	The system time window Tsw for post initiator HRAs was frequently associated with 'core damage'. Post initiator HRAs that appear in the top cutsets may require success criteria linked to beginning of core uncovery (about 20 minutes before 'core damage'). Or, the operator actions that may fall into that final 20-minute time period should be overridden to assume a high stress level. While section 3.1.5.7 described this approach, there is no evidence of its proper application in the HRA quantifications. (This F&O originated from SR HR-H2)	Complete	It was determined that the text in Section 3.1.5.7 was incorrect and does not capture how stress is actually applied in the EPRI HRA Calculator. C0-HR-001, Internal Events PRA Human Reliability Analysis, has been updated to show the stress level applied to each HFE and the justification for stress selection. Also included is a correlation between stress level and failure of execution probability. New text has been provided for inclusion in a future update of the HRA notebook. REFERENCE C0-HR-001 C0-HRA-001	No impact. As described in this F&O for internal events, the stress levels in the model are appropriate, but updates to the documentation are required. The internal events documentation was updated.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
6-22	HR-E1	Human Reliability	Upon RAS, LPSI stops and EOP-5, Step S.1(d) requires the Operators to 'Shut RWT OUT Valves SI-4142, 4143'. This manual action was not modeled in the PRA. The CCNPP PRA staff provided reasonable response to this issue. Based on CR-2009-005581, there is no impact on pump operability. Also, the staff will continue to track the CR. If there are any changes to the disposition of pump operability, then a new HRA may be added to the PRA model (if warranted). (This F&O originated from SR HR-E1)	Complete	As documented in CR-2009-005881, shutting the RWT outlet valves upon a RAS does not impact station operability. The Safety Injection Pumps and Containment Spray Pumps will not fail if the RWT isolation valves do not closed with a RAS signal. A design margin issue has been identified. This issue has been added to the plant's margin management program. No model changes have been made, but the PRA configuration management program, CNG-CM-1.01-3003, would capture any design changes concerning this issue. REFERENCES C0-SY-052 CR-2009-005881 CNG-CM-1.01-3003	No impact. The system is operable without the manual action to shut the RWT outlet valves. There is no impact on internal events CDF/LERF. The issue was added to the plant's margin management program.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
6-23	HR-G7	Human Reliability	When the Calculator reads in the combinations, it assumes that actions occur in the order of the time delay (Td). However, the time delay is not the same for all sequences, and care must be taken to make the combinations appropriate for the sequences in which they occur. Page 88 of the HRA notebook indicates this was considered, since the Td was modified for events occurring prior to reactor trip, and also for OTCC after SG overfill. However, not all occurrences have been addressed. The combination examined by the review team is Combination 770 (OTCC after CST depletion). In this event the CST depletion should come first. (This F&O originated from SR HR-G7)	Open	New HRA events, CVC0HFOTA8HRS and AFW0HFCCSGDEC8HR were added to model Td variances where CST depletion occurs early and when it occurs later. This account for appropriate sequencing of events. An updated dependency analysis has been performed, which includes these new HRA events. The dependency analysis shows that these new HRA actions are not significant for CDF or LERF. A PRA configuration control database (CRMP) item has been initiated to formally incorporate the updated dependency analysis into the model. REFERENCES C0-HR-001 C0-HRA-001	This specific issue with time delay and CST depletion has been addressed and incorporated into the PRA model.

Table A-1 Internal Events PRA Peer Review – Facts and Observations

F&O ID	SR	Topic	Finding/Observation	Status	Disposition	Impact to CDF/LERF
7-13	QU-A2	Quantification	Discrepancy between documentation and result files. SBO037 and SBO038 sequences appear to be inverted in Tables D-1, 4.2.2, 4.2.4, 4.2.5, B-3). (This F&O originated from SR QU-A2)-	Complete	The top flood cutset was incorrectly flagged as being SBO sequence 37 (offsite power recovered < 1 hour) instead of sequence 38 (offsite power not recovered). Updated tables B-2, C-1, and D 1 in C0-QU-001. Spot-check was performed to identify other errors. In C0-QU-002, fixed sequence 12 table 4.2-5, which incorrectly showed sequence 37 instead of 38. REFERENCES C0-QU-001 C0-QU-002	No impact. This was an Internal events documentation issue.

B. Fire PRA Quality (27 Pages)

In accordance with RG 1.205 position 4.3:

"The licensee should submit the documentation described in Section 4.2 of Regulatory Guide 1.200 to address the baseline PRA and application-specific analyses. For PRA Standard "supporting requirements" important to the NFPA 805 risk assessments, the NRC position is that Capability Category II is generally acceptable. Licensees should justify use of Capability Category I for specific supporting requirements in their NFPA 805 risk assessments, if they contend that it is adequate for the application. Licensees should also evaluate whether portions of the PRA need to meet Capability Category III, as described in the PRA Standard."

The CCNPP FPRA peer review was performed January 16-20, 2012 using the NEI 07-12 Fire PRA peer review process, the ASME PRA Standard (ASME/ANS RA-Sa-2009) and Regulatory Guide 1.200, Rev. 2. The purpose of this review was to establish the technical adequacy of the FPRA for the spectrum of potential risk-informed plant licensing applications for which the FPRA may be used. The 2012 Calvert FPRA peer review was a full-scope review of all of the technical elements of the CCNPP at-power FPRA (2012 model of record) against all technical elements in Section 4 of the ASME/ANS Combined PRA Standard, including the referenced internal events SRs. The peer review noted a number of facts and observations (F&Os). The findings and their dispositions are provided in Table B-1. All findings are being provided and have been dispositioned. All F&Os that were defined as suggestions have been dispositioned and will be available for NRC review. The FPRA is adequate to support the NFPA 805 licensing basis.

The FPRA now meets at least Capability Category II in all cases. Eleven ASME/ANS SRs were identified by the peer review team as meeting Capability Category I only requirements or a level of "not met" for the requirement. The capability categories are defined in ASME/ANS RA-Sa-2009. An evaluation of the impact of those areas where only the Capability Category I requirement was met or the requirement was "not met" is provided in Table B-2 along with the basis for now meeting Capability Category II.

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
PP-B3-01	PP-B3 PP-B6 PP-C3	Plant Partitioning	Complete	The containment is partitioned into 2 PAUs. There are intervening combustibles and this was accounted for in the PRA by treating the 20 feet as an overlap region and failing components affected in both PAUs. There is no justification given for the 20 foot assumption. The turbine deck is continuous from unit 1 to unit 2. This area is divided into 2 PAUs, TURB1 and TURB2, but there is no discussion for the basis of the partitioning. Finding level of significance is based on crediting spatial separation with no requisite justification. Maintain the containment as 1 PAU and discern the separation of east from west in the fire modeling. Document the spatial separation and no intervening combustibles for the turbine deck.	C0-PP-001, Calvert Cliffs Fire PRA Plant Partitioning Notebook, was updated to include an analysis that justifies the partitioning of the containment into two plant partitioning units with a 20-foot spatial separation (known as the buffer zone). The only potential intervening combustibles in this buffer zone were identified as qualified cables that were verified to be encased within marinate covered raceways. The covers prevent the cables from becoming potential combustibles and therefore are not considered intervening combustibles. The unit 1 and unit 2 Turbine Deck was walked down to assess for the acceptability of the Appendix R partitioning into distinct PAUs. The boundary was assessed to have at least a 20-foot separation between potential ignition sources and potential targets, assessed for intervening combustibles, and the Turbine deck volume assessed for damaging hot gas layer development. The partitioning was found acceptable and consistent with NUREG/CR-6850, Section 1.5.2, where main turbine decks are typical applications where spatial separation has been credited.

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
PP-B5-01	PP-B5 PP-C3	Plant Partitioning	Complete	The water curtain in the CCW room was credited as an active fire barrier. The justification was that the water curtain was part of the original regulatory fire protection program. This meets CAT 1, but needs enhancement for CAT II/III. Finding level was used because the requirements for CAT II/III were not met. Calvert Cliffs should provide a direct reference to their Appendix R program as the basis for the acceptability for this or provide a design basis justification for the water curtain and document that in the PP notebook if the Appendix R program reference cannot be found.	The Component Cooling Water room water curtain is an approved Appendix R exemption, as identified in the exemption issued by the NRC in response to Calvert Cliffs exemption request ER820816. The validity of crediting CCW Room Water Curtains is discussed in Southwest Research Institute Report No. 01-0763-201. A reference to the Southwest Research Institute report was added to C0-PP-001, Plant Partitioning Notebook.

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings					
F&O ID	**SR**	**Topic**	**Status**	**Finding**	**Disposition**
PP-B7-01	PP-B7 PP-C3 PP-C4 QLS-A1	Plant Partitioning Qualitative Screening	Complete	1. The walk down nomenclature does not match the PP notebook. Example page 561 of the walkdown documentation uses nomenclature in the containment that does not match the PP notebook. 2. There are many areas inaccessible such as: #23 Charging Pump Room, U1 Service Water Pump Room, U1 East Battery Room, E/W Corridor. These areas appear to be accessible with a little effort. In some of the areas screened out in QLS, the areas were inaccessible and did not have a confirmatory walkdown. Finding level assessed due to the incompleteness of the walkdown documentation. 1. Prepare a table that correlates the PAUs from the PP notebook with the area nomenclature used in the walkdown documentation. 2. Complete the walkdowns, particularly for areas screened in the QLS task.	A table was created to correlate the building or area nomenclature that was used for the plant walkdown documentation, to the plant analysis unit identifiers used in the Fire PRA analysis. This table was added to C0-PP-001, Calvert Cliffs Fire PRA Plant Partitioning Notebook as Table 17. The facilities and rooms that were not originally walked-down were reviewed. Supplemental walkdowns were performed and supplemental walkdown datasheets were generated. For areas that were not accessible at the time of the supplemental walkdowns (for radiological safety reasons, personnel safety concerns, or access otherwise denied), The reason for inaccessibility was added to Table 17.

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F&O ID	SR	Topic	Status	Finding	Disposition
CS-B1-01	CS-B1 CS-C4	Fire PRA Cable Selection and Location	Complete	Current Breaker coordination study still in progress. This study needs to be completed in order to receive a category II met for CS-B1. Complete the breaker coordination study.	The breaker coordination study has been completed. As described in ECP-13-000321, Form 12, Engineering Evaluation, all PRA common power supplies are assumed to meet - or will meet - the coordination requirements of NFPA 805, except as noted in C0-CS-001, Fire PRA Cable Selection Notebook. As described in the cable selection notebook, two 120VAC lighting panels are not validated as coordinated, and these panels are assumed to fail for all Fire PRA scenarios. Also, as described in the PRA notebook a breaker for 480V motor control center MCC101BT has not been validated as coordinated. This breaker, 52-10150, is modeled so that a fire-induced electrical fault on the breaker's power cabling will fail MCC101BT. Finally, the notebook identifies that selected 120V power panels have coordination issues, but that these will be addressed by design changes and referenced in Attachment S – Modifications and Implementation Items.
PRM-B3-01	PRM-B3 PRM-B4 PRM-B5	Fire PRA/Plant Response Model	Complete	The FPRA model did not address events involving loss of both HVAC trains to the MCR, long term heatup of MCR and need for operator actions outside the MCR to compensate for the loss of electronic controls in the MCR, which was assumed as a CCDP of 1.0 for the plant. The basis for excluding this potential Core Damage sequence was addressed in questions to the Calvert Cliffs PRA team. This sequence is a new sequence outside the current model FPRA model logic trees.	Loss of Control Room HVAC can affect the operability and availability of equipment in the control room and cable spreading room. As described in Calvert PRA System Analysis Notebooks C0-SY-002, C0-SY-017, and C0-SY-030, loss of HVAC is modeled to have the effect of increasing the failure rate of 120VAC and 125VDC instruments and controls in the cable spreading room. For the control room, degradation of the 125VDC system is used as a conservative surrogate for control room I&C degradation. Loss of Control Room HVAC and subsequent temperature increases may adversely affect

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
				Consider using a combination of MCR heatup calculations to define the time when operators would leave MCR and consider a recovery action for restoring cooling the MCR.	operator responses. The model reflects degradation of human actions by the degradation of the 125VDC system used for instruments and controls. Loss of Control Room HVAC is not expected to cause abandonment by operations staff of the control room due to high temperatures. On complete loss of HVAC with no mitigation, such as no use of emergency fans, calculation CA02725 shows a CR temperature of 123 deg F at 24-hours. While this is a challenging environment, this temperature is assessed as insufficient to solely drive a complete CR abandonment scenario. NUREG/CR-6738 describes operational experience where operators will continue to occupy the control room even under severe environments. Operations staff says that in consideration of high temperatures in the control room, that Operations would do what was needed to keep the cores safe and covered. The site safety director says that for a temperature of 123 deg F, the site would implement a mitigation strategy which would include stay-times, assessment of individuals for heat-related conditions, use of ice vests, and call-in of additional qualified operations staff to rotate into the control room. The above discussion was included in C0-SY-030, Control Room HVAC PRA System Notebook.
FSS-A5-01	FSS-A5	Fire Scenario Selection and Analysis	Complete	A range of ignition source / target set combinations has been represented for unscreened PAUs. These combinations are identified in relevant calculation sheets for unscreened PAUs. In some	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. In both cases, thermocouple location was adjusted as identified in F&O FSS-D3-02. For the CSR, consequences were

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
				PAUs, sub-PAUs are defined and damage from a potential fire within the sub-PAU is addressed. However, it is not clear how or why damage would be limited to the specified sub-PAU because there are no physical barriers between specified sub-PAUs. The documentation is such that it cannot be determined if the selected fire scenarios provide reasonable assurance that the risk contribution of each unscreened PAU can be characterized. Another issue that influences the potential for fire propagation across sub-PAU boundaries is that the temperature measurement locations specified in the detailed FDS fire modeling evaluations do not generally coincide with locations where maximum temperature are expected (e.g., within the fire plume). As a consequence, for some fire scenarios damage to targets is not predicted when it should be based on the specified damage criteria. Some scenarios are screened on the basis of temperature measurements that do not represent conditions at targets within the fire plume. (See F&O FSS-D3-02) This could have a significant impact on the potential for fire propagation across sub-PAU boundaries and needs to be discussed more thoroughly.	divided into scenarios based on mitigation potential. First, if the scenario was suppressed by the Halon system then the limit of damage was based on what was predicted by FDS in terms of temperature and energy. If it was unsuppressed it went to total room burn, which assumes failure of all targets in the room, regardless of the initial scenario boundary. For the Switchgear Room FDS analysis, the analysis was updated to add clarity to the analysis. A discussion of the application of sub-PAUs has been added to Addendum 1 to C0-FSS-004, Fire PRA Detailed Fire Modeling Notebook. Damage was not limited to specified sub-PAUs. Specific examples of the treatment of fire growth and the application of sub-PAUs have been provided. As described in C0-FSS-004, the sub-PAU analysis included spatial information from walkdown, along with engineering judgment, to determine if fire sources could fail additional components, cables, or other combustibles, potentially leading to more damage to surrounding equipment or cables. For scenarios that leveraged FDT modeling, the issue related to whether the analysis had correctly addressed the impact of transients along the edge of a boundary interface for a sub-PAU. A comparable consideration was also related to secondary combustion and oil fires. Resolution involved selection of several representative PAUs for a sensitivity study that expanded the existing sub-PAUs and examined secondary ignition potential.

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
FSS-A5-01	FSS-A5	Fire Scenario Selection and Analysis	Complete	There were indications that Calvert Cliffs had the tools and information in place to properly evaluate the propagation of fires across the sub-PAU boundaries given no physical barriers but there were no examples showing that this evaluation was performed or any explicit descriptions of how they were performed in general. The concern here is that without an explicit description of the process for evaluating the spread of fires across sub-PAU boundaries with no physical barriers and detailed examples, there is the potential that in the future, new people updating the PRA may not know that they have to evaluate this. Calvert Cliffs needs to describe their process for evaluating fire growth and propagation between sub-PAUs and as applicable, between PAUs. Specific examples of the sub-PAU fire growth need to be provided. If fire propagation from sub-PAU to sub-PAU was not treated, Calvert Cliffs needs to evaluate all sub-PAUs to determine if there is any potential for fire spread and then model the potential for spreading fires and for damage occurring across sub-PAU boundaries.	The PAUs were considered representative of the work performed based on several criteria. The analysis indicated that the methods mentioned were indeed appropriate. Sub-PAU impacts did not change from the expanded assessment and that secondary ignition was bounded by the existing analysis and was appropriately addressed. The analysis was incorporated into the documentation for C0-FSS-004.

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
FSS-D2-01	FSS-D2	Fire Scenario Selection and Analysis	Complete	Where used, the FDS model was generally used with a level of grid resolution that was below the level of grid resolution documented in the NUREG-1824 Verification and Validation study for the FDS model. A validation study was not conducted to support the use of this lower level of grid resolution. Grid resolution has a bearing on the results of FDS calculations. Grid resolutions outside the validation range in NUREG-1824 should be justified and validated. Increase the level of grid resolution in the FDS PAU Fire Evaluations (C0-FSS-004 R1) so that the grid resolution is within the validation range documented in NUREG-1824.	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. For the Cable Spreading Room FDS fire scenarios, a grid study was performed on the updated FDS model. The study recommended a grid size that was within the range in NUREG/CR-1824. That grid size was used for CSR FDS scenario evaluations. The study and results were incorporated into C0-FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' Switchgear Rooms were updated to increase the level of grid resolution to a value that is within the validation range documented in NUREG/CR-1824. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1.

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
FSS-D3-01	FSS-D3 FSS-B2 FSS-D4	Fire Scenario Selection and Analysis	Complete	This SR is not met because detailed FDS fire modeling evaluations of PAUs 302, 306, 311, 317, 407 and 430 assume that material surfaces are "inert." As noted on p. 44 of C0-FSS-004 R1, this assumption was made "... so that no objects in the PAU or the PAU structure (walls, floor, or ceiling) itself would absorb any heat from the various fire scenarios, producing a more conservative or worst case result for all fire scenarios' impacts to the components and cables within the PAU model. As such, no detailed material properties were required to be defined in FDS for the scenarios to function correctly." However, specification of material surfaces as "inert" in FDS does not prevent heat absorption into material surfaces. On the contrary, this specification maintains material surfaces at ambient temperature in FDS, which tends to maximize heat absorption into these surfaces.	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. For the Cable Spreading Room FDS fire scenarios, the Unit 1 CSR was modified to include actual material properties and sensitivity analysis. Actual material properties were used in the updated U1CSR FDS model rather than the prior use of "inert" material conditions. Adiabatic conditions were used for any items with material properties that are unknown or of a high uncertainty to bound the analysis and prevent heat transfer into those objects. The CSR FDS model was executed and the results compared to the baseline results. This study was then documented in FSS-004. The results were applied to Unit 2 CSR. This study was then documented in FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' Switchgear Rooms were updated to specify representative material properties as referenced by NUREG 1805. This adjustment enabled the analysis to obtain more realistic estimates of environmental conditions for these fire scenarios. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1.

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F&O ID	SR	Topic	Status	Finding	Disposition
FSS-D3-01	FSS-D3 FSS-B2 FSS-D4	Fire Scenario Selection and Analysis	Complete	To prevent heat absorption into material surfaces, they should have been specified as "adiabatic" rather than as "inert." The "inert" parameter in FDS maximizes heat transfer to surfaces rather than minimize it. This can result in lower calculated gas temperatures. Specify materials surfaces as "adiabatic" rather than as "inert" in FDS to prevent them from absorbing heat in order to achieve the stated goal of producing a more conservative or worst case result. This may prove to be overly conservative, in which case specification of realistic material properties could be used to achieve more realistic estimates of environmental conditions for these fire scenarios.	

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F&O ID	**SR**	**Topic**	**Status**	**Finding**	**Disposition**
FSS-D3-02	FSS-D3 FSS-A5	Fire Scenario Selection and Analysis	Complete	Temperature measurement locations specified in the detailed FDS fire modeling evaluations do not generally coincide with locations where maximum temperature are expected (e.g., within the fire plume). As a consequence, for some fire scenarios damage to targets is not predicted when it should be based on the specified damage criteria. Some scenarios are screened on the basis of temperature measurements that do not represent conditions at targets within the fire plume. Re-run FDS simulations with temperature measurement probes located within the fire plume or use other fire modeling tools such as FDTs to calculate fire plume temperatures for these scenarios.	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. For the Cable Spreading Room FDS fire scenarios, new measurement devices were included in the updated U1CSR FDS model. The thermocouples were placed directly above the fire source in the updated FDS model and the scenarios re-evaluated. The results were applied to Unit 2 CSR. This study and the results were then documented in FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' SWGR rooms were updated to alter the location of the thermocouples such that the centerline plume temperature was recorded and used to determine target impacts. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1.

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
FSS-D8-01	FSS-D8	Fire Scenario Selection and Analysis	Complete	Fire detection timing is evaluated for detailed fire modeling cases that use FDS. This fire detection timing is then used to estimate automatic fire suppression timing and fire brigade response timing for these scenarios. However, the fire detection timing is based on modeling that does not include obstructions located beneath the ceiling that could have an impact on fire detector response. The fire detection timing is also based on an unjustified assumption regarding the type of smoke detectors installed in the affected PAUs. Obstructions to the flow of fire gases can have an impact on smoke concentrations and velocities, which in turn influence smoke detector response. Without including such obstructions in fire modeling simulations, their impact on fire detection times is not evaluated. Include obstructions located beneath the ceiling for the affected fire scenarios in order to evaluate their impact on fire detection timing. Provide justification for the selection of the type of smoke detector specified in the FDS simulations for these fire scenarios.	FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. For the updated Cable Spreading Room FDS fire scenarios, cable tray obstructions were placed in the ceiling area of the updated U1CSR FDS model. Additional thermocouple and heat flux data recording devices were added to the U1CSR model under the new cable tray obstructions in the vicinity of the fire source. The scenarios were re-evaluated. The results were applied to Unit 2. A sensitivity study was also performed. The study and new scenario results were incorporated into C0-FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' SWGR rooms were also updated to include significant obstructions such as cable trays and beam pockets within the switchgear rooms. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results and details of this analysis are documented in C0-FSS-004 as Addendum 1.

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
FSS-F3-01	FSS-F3	Fire Scenario Selection and Analysis	Complete	To achieve CC II/III for this SR, a quantitative assessment of the risk of the selected fire scenarios involving a) exposed structural steel and b) the presence of a high-hazard fire sources must be completed consistent with the FQ requirements including the collapse of the exposed structural steel and any attendant damage. Such an assessment has not been done or was not documented in a readily discernible manner. This has a potential impact on fire risk quantification. Complete a quantitative assessment of the risk of the selected exposed structural steel fire scenarios consistent with the FQ requirements.	The Turbine Building was reviewed for potential fire scenarios where structural steel can be adversely affected. From the scenarios examined, those that can damage structural steel were selected for further analysis. The frequency, severity factor and non-suppression probability of each scenario were developed and included in the Structural Failure Analysis Notebook. These impacts were then added to FRANX database and quantified as part of the final Fire PRA risk quantification in Fire Quantification Notebooks C0-FRQ-001 and C0-FRQ-002.
FSS-G4-01	FSS-G4	Fire Scenario Selection and Analysis	Complete	An assessment of the effectiveness, reliability and availability of credited passive fire barrier features has not been documented in the multi-compartment analysis. To achieve a CC II capability assessment, the effectiveness, reliability and availability of credited passive fire barrier features must be assessed. Assess the effectiveness, reliability and availability of credited passive fire barrier features and document this assessment.	Generic probabilities were used for credited passive fire barrier features in the multi-compartment analysis. At Calvert Cliffs, the fire barriers are verified to be effective through test procedures. An unreliability value was applied to all normally closed doors that represents the probability of the door being propped open given a fire in the exposing compartment. The probability of finding a failed sealed wall penetration is assumed to be very small to warrant propagation scenarios. A discussion of the effectiveness, reliability, and availability of fire barriers was added to C0-FSS-008, Calvert Fire PRA Multi-Compartment Analysis.

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F&O ID	SR	Topic	Status	Finding	Disposition
FSS-G5-01	FSS-G5	Fire Scenario Selection and Analysis	Complete	The effectiveness, reliability and availability of credited active fire barrier features have not been quantified in the multi-compartment analysis. To achieve a CC II capability assessment, the effectiveness, reliability and availability of credited active fire barrier features must be quantified. Quantify the effectiveness, reliability and availability of credited active fire barrier features and document this assessment.	Active fire barriers were evaluated as effective in studies used to support Appendix R analysis. An unreliability value has been applied to all normally open, self closing dampers and doors; A discussion of the effectiveness of credited active fire barriers was added to C0-FSS-008, Calvert Fire PRA Multi-Compartment Analysis.
HRA-B2-01	HRA-B2	Human Reliability Analysis	Complete	Improve documentation of the adverse operator actions needed to address the impact of grounded or shorted electrical buses that might have an impact on other plant buses if not isolated and re energized in the areas identified. Very difficult to find the information within the HRA notebook alone, because the actions are modeled as inputs to FRANX. Provide new tables listing the actions considered or references to specific locations.	C0-HRA-001, Fire Human Reliability notebook, was updated to detail the adverse operator actions added to the model following the fire AOP review process. Table 3 was added to Section 2.2 detailing each basic event, set to true (1.0) used in the model to annotate the adverse operator actions in the model. These include actions to de-energize electrical busses to isolate them from potential shorts and grounds. Table 2 shows the HFEs added to the model as part of the AOP review, including actions to restore AC power to busses lost due to fire failure sequences.
HRA-E1-01	HRA-E1	Human Reliability Analysis	Complete	Documentation for what was done was very good, however, the details for not selecting any spurious alarms is not clear. The documentation of the adverse actions put into the model as "true" are not in the HRA report, actions identified in the cutset reviews are not	C0-HRA-001, Fire Human Reliability Notebook, was updated detailing the Alarm Response Procedure review process. Table 12 was expanded to show the ARP review of alarm impact and operator interview notes for CR annunciators that could result in a manual reactor trip. No annunciators were identified that would cause the

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
				clearly identified, rational for not using specific HFEs in the RCP trip actions, for identifying actions from procedures and the process for assigning uncertainty range for the combos. Doesn't permit verification of the rational for judgments made in deciding what is in and out of the Fire HRA. Also, from the calculation viewpoint the need to know the use of all manpower requirements during early time after fire initiator for dependency analysis. Enhance documentation of the specific issues needed to reproduce the assumptions and calculations used in the HRA.	operator to terminate a systems or components operation based solely on the alarm itself, but several were identified that could potentially result in the operator tripping the Unit unnecessarily. C0-HRA-001 was also updated to detail the adverse operator actions added to the model following the fire AOP review process. Table 3 was added to Section 2.2 detailing each basic event, set to true (1.0) used in the model to annotate the adverse operator actions in the model. These include actions to de-energize electrical busses to isolate them from potential shorts and grounds. Table 2 shows the HFEs added to the model as part of the AOP review, including actions to restore AC power to busses lost due to fire failure sequences. New HFEs added as part of the cutset review process are identified in Table 1 of C0-HRA-001, Fire Human Action Reliability notebook. These are annotated with "identified during the development of the PRM Notebook." The cutset reviews are described in C0-QNS-001, Fire PRA Quantitative Screening Notebook. A new dependency analysis was performed after the new HFEs were added to the model, ensuring new dependency combinations are considered. Additional information was added to Table 1 of the Human Reliability Analysis Notebook, C0-HRA-001, detailing why each HFE was either retained or removed. For example, event FGAFW0SGTRISOL, Operator Feeds Affected SG with SGTR to Assure Heat Removal, was "Not

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F&O ID	SR	Topic	Status	Finding	Disposition
					retained for fire scenarios, because these actions are SGTR specific. Modeling was not necessary to ensure these actions did not appear in the cut sets, because the SGTR initiator is not being used for fire scenarios." Combination event multipliers are used in cutsets of multiple HEP actions to account for dependencies between HEP actions. To account for the uncertainty in HEP actions, an uncertainty parameter is added to the HEP action. When performing uncertainty analysis, the uncertainty parameters for combination events is increased proportionally when they are multiplied by the combination event multipliers. Based on interviews, there are sufficient non-control room personnel for fire recovery actions. Appendix D of C0-HRA-001 notes that there are no control room operators assigned to the fire brigade. There were no identified staffing issues or interferences between operators performing fire recovery actions and members of the fire brigade.
FQ-A1-01	FQ-A1	Fire Risk Quantification	Complete	Treatment of 0 CCDPs scenarios is not clear and appears to result in an underestimate of total risk (the underestimate appears to be small based on the sensitivity evaluations performed): 1 - with respect to opposite unit quantification, use CCDP for reactor trip initiator unless confirmation of no trip is documented; 2 - address use of 0 CCDP for control	The fire risk quantification process has been updated in notebooks C0-FRQ-001 and C0-FRQ-002 to address the issue with FRANX fire scenarios having a zero conditional probability for CDF and LERF. 1. When documented analysis shows that selected fire scenarios for one unit are screened from impact for the opposite unit (typically, no trip would be initiated), then that scenario may be excluded from the opposite unit's fire risk quantification.

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
				room HVAC loss scenarios, apply CCDP consistent with control room abandonment 3 - for scenarios with limited impact with a 0 CCDP, due to cutsets below truncation limit, apply a baseline CCDP based on reactor trip initiator More than 50% of the scenarios have a 0 CCDP but no clear discussion of the basis for the 0 CCDP is provided. Treatment of 0 CCDPs scenarios: 1 - with respect to opposite unit quantification, use CCDP for reactor trip initiator unless confirmation of no trip is documented; 2 - address use of 0 CCDP for control room HVAC loss scenarios, apply CCDP consistent with control room abandonment 3 - for scenarios with limited impact with a 0 CCDP, due to cutsets below truncation limit, apply a baseline CCDP based on reactor trip initiator	Otherwise, a nominal conditional probability, as described in item 3 below, would apply. 2. F&O PRM-B3-01 identifies the concern with loss of Control Room HVAC with control room abandonment. As discussed in more detail with the resolution to PRM-B3-01, subsequent investigation revealed that loss of CR HVAC is not expected to cause abandonment by the operations staff of the control room due to high temperatures. Loss of CR HVAC and subsequent temperature increases may adversely affect operator responses, and the model reflects degradation of human actions with loss of CR HVAC. C0-SY-030, Control Room HVAC PRA System Notebook, was updated to include this discussion. 3. The new quantification process described in the FRQ notebooks is to assure a nominal conditional value is calculated for these low significant scenarios by 1) recalculating the zero-conditional scenarios at a lower truncation value to assure resolution in the scenario cutset file and conditional probabilities , and/or to 2) use a baseline conditional probability for CDF and LERF for the internal events reactor trip initiating vent - IE0PT for Unit 1 or IE0PT-2 for Unit 2
FQ-B1-01	FQ-B1	Fire Risk Quantification	Complete	We observed zero CCDPs for some PAU CDF and LERF values in the FRANX tables (e.g., PAU 512) which eliminated loss of HVAC to the MCR as a potential MCR abandonment sequence. Treatment of 0 CCDPs scenarios:	The fire risk quantification process has been updated in notebooks C0-FRQ-001 and C0-FRQ-002 to address the issue with FRANX fire scenarios having a zero conditional probability for CDF and LERF. 1. When documented analysis shows that selected

Table B-1 Fire PRA Peer Review – Facts and Observations – Findings

F&O ID	SR	Topic	Status	Finding	Disposition
				1 - with respect to opposite unit quantification, use CCDP for reactor trip initiator unless confirmation of no trip is documented; 2 - address use of 0 CCDP for control room HVAC loss scenarios, apply CCDP consistent with control room abandonment (F&O FQ-A1-01 (F)) 3 - for scenarios with limited impact with a 0 CCDP, due to cutsets below truncation limit, apply a baseline CCDP based on reactor trip initiator Allowing zero CCDPs allows scenarios in the fire model to quantify with no contribution to the CDF or LERF value and this under represents those frequencies especially when considering delta risk evaluations. Replace the zero entries with the lowest CCPD for a plant trip with only random failures of the safety equipment as in the internal events model. We discussed this with the Calvert Cliffs PRA team and some of the zeros are due to fire areas in one unit potentially contributing to the CCDP of the opposite unit. With the exception of these cases a method for handling the zeros needed to be developed and applied in the frequency quantifications.	fire scenarios for one unit are screened from impact for the opposite unit (typically, no trip would be initiated), then that scenario may be excluded from the opposite unit's fire risk quantification. Otherwise, a nominal conditional probability, as described in item 3 below, would apply. 2. F&O PRM-B3-01 identifies the concern with loss of Control Room HVAC with control room abandonment. As discussed in more detail with the resolution to PRM-B3-01, subsequent investigation revealed that loss of CR HVAC is not expected to cause abandonment by the operations staff of the control room due to high temperatures. Loss of CR HVAC and subsequent temperature increases may adversely affect operator responses, and the model reflects degradation of human actions with loss of CR HVAC. C0-SY-030, Control Room HVAC PRA System Notebook, was updated to include this discussion. 3. The new quantification process described in the FRQ notebooks is to assure a nominal conditional value is calculated for these low significant scenarios by 1) recalculating the zero-conditional scenarios at a lower truncation value to assure resolution in the scenario cutset file and conditional probabilities , and/or to 2) use a baseline conditional probability for CDF and LERF for the internal events reactor trip initiating vent - IE0PT for Unit 1 or IE0PT-2 for Unit 2

Table B-2 Fire PRA – Category I Summary

SR	Topic	Status
PP-B3	2012 Peer Review: SR Not Met The containment is partitioned into 2 PAUs. There are intervening combustibles and this was accounted for in the PRA by treating the 20 feet as an overlap region and failing components affected into both PAUs. There is no justification given for the 20 assumption. The turbine deck is continuous from unit 1 to unit 2. This area is divided into 2 PAUs, TURB1 and TURB2, but there is no discussion for the basis of the partitioning. Associated F&O: PP-B3-01	Now: Met Cat II/III C0-PP-001, Calvert Cliffs Fire PRA Plant Partitioning Notebook, was updated to include an analysis that justifies the partitioning of the containment into two plant partitioning units with a 20-foot spatial separation (known as the buffer zone). The only potential intervening combustibles in this buffer zone were identified as qualified cables that were verified to be encased within marinate covered raceways. The covers prevent the cables from becoming potential combustibles and therefore are not considered intervening combustibles. The unit 1 and unit 2 Turbine Deck was walked down to assess for the acceptability of the Appendix R partitioning into distinct PAUs. The boundary was assessed to have at least a 20-foot separation between potential ignition sources and potential targets, assessed for intervening combustibles, and the Turbine deck volume assessed for damaging hot gas layer development. The partitioning was found acceptable and consistent with NUREG/CR-6850, Section 1.5.2, where main turbine decks are typical applications where spatial separation have been credited.
PP-B5	2012 Peer Review: SR Met: (CC-I) The water curtain in the CCW room was credited as an active fire barrier. The justification was that the water curtain was part of the original regulatory fire protection program. This meets CAT 1, but needs enhancement for CAT II/III Associated F&O: PP-B5-01	Now: Met Cat II/III The Component Cooling Water room water curtain is an approved Appendix R exemption, as identified in the exemption issued by the NRC in response to Calvert Cliffs exemption request ER820816. The validity of crediting CCW Room Water Curtains is discussed in Southwest Research Institute Report No. 01-0763-201. A reference to the Southwest Research Institute report was added to C0-PP-001, Plant Partitioning Notebook.

Table B-2 Fire PRA – Category I Summary

SR	Topic	Status
PP-B6	2012 Peer Review: SR Not Met The containment has a 20 foot area that overlaps between the E and W section. The overlap is specifically addressed in the PP notebook. The standard does not allow for an overlap. Associated F&O: PP-B3-01	Now: Met Cat I/II/III C0-PP-001, Calvert Cliffs Fire PRA Plant Partitioning Notebook, was updated to include an analysis that justifies the partitioning of the containment into two plant partitioning units with a 20-foot spatial separation (known as the buffer zone). The only potential intervening combustibles in this buffer zone were identified as qualified cables that were verified to be encased within marinate covered raceways. The covers prevent the cables from becoming potential combustibles and therefore are not considered intervening combustibles.
CS-B1	2012 Peer Review: SR Met: (CC I) Supporting Requirement CS-B1 met with a category I. A breaker coordination study is currently being performed and is planned to be incorporated in the future. See Fact and Observation CS-B1-01. Associated F&O: CS-B1-01	Now: Met Cat II/III The breaker coordination study has been completed. As described in ECP-13-000321, Form 12, Engineering Evaluation, all PRA common power supplies are assumed to meet - or will meet - the coordination requirements of NFPA 805, except as noted in C0-CS-001, Fire PRA Cable Selection Notebook. As described in the cable selection notebook, two 120VAC lighting panels are not validated as coordinated, and these panels are assumed to fail for all Fire PRA scenarios. Also, as described in the PRA notebook a breaker for 480V motor control center MCC101BT has not been validated as coordinated. This breaker, 52-10150, is modeled so that a fire-induced electrical fault on the breaker's power cabling will fail MCC101BT. Finally, the notebook identifies that selected 120V power panels have coordination issues, but that these will be addressed by design changes and referenced in Attachment S – Modifications and Implementation Items.

Table B-2 Fire PRA – Category I Summary

SR	Topic	Status
PRM-B3	2012 Peer Review: SR Not Met No new initiating events were identified in the course of the fire PRA model generation. The failure of the control room HVAC does not lead to a control room abandonment CCDP (1.0 or other value justified by analysis as corresponding to shutdown from outside the control room) Associated F&O: PRM-B3-01	Now: Met Cat I/II/III Loss of Control Room HVAC can affect the operability and availability of equipment in the control room and cable spreading room. As described in Calvert PRA System Analysis Notebooks C0-SY-002, C0-SY-017, and C0-SY-030, loss of HVAC is modeled to have the effect of increasing the failure rate of 120VAC and 125VDC instruments and controls in the cable spreading room. For the control room, degradation of the 125VDC system is used as a conservative surrogate for control room I&C degradation. Loss of Control Room HVAC and subsequent temperature increases may adversely affect operator responses. The model reflects degradation of human actions by the degradation of the 125VDC system used for instruments and controls. Loss of Control Room HVAC is not expected to cause abandonment by operations staff of the control room due to high temperatures. On complete loss of HVAC with no mitigation, such as no use of emergency fans, calculation CA02725 shows a CR temperature of 123 deg F at 24-hours. While this is a challenging environment, this temperature is assessed as insufficient to solely drive a complete CR abandonment scenario. NUREG/CR-6738 describes operational experience where operators will continue to occupy the control room even under severe environments. Operations staff says that in consideration of high temperatures in the control room, that Operations would do what was needed to keep the cores safe and covered. The site safety director says that for a temperature of 123 deg F, the site would implement a mitigation strategy which would include stay-times, assessment of individuals for heat-related conditions, use of ice vests, and call-in of additional qualified operations staff to rotate into the control room. The above discussion was included in C0-SY-030, Control Room HVAC PRA System Notebook.

Table B-2 Fire PRA – Category I Summary

SR	Topic	Status
PRM-B4	2012 Peer Review: SR Not Met See PRM-B3-01 F&O, not met due to no new initiators identified and the identification of a potential new initiator that was not quantified in the fire PRA model. Associated F&O: PRM-B3-01	Now: Met Cat I/II/III The potential new initiator has been assessed (failure of CR HVAC leading to CR abandonment as discussed in PRM-B3). Loss of Control Room HVAC and subsequent temperature increases may adversely affect operator responses. The model reflects degradation of human actions by the degradation of the 125VDC system used for instruments and controls. Loss of Control Room HVAC is not expected to cause abandonment by operations staff of the control room due to high temperatures. On complete loss of HVAC with no mitigation, such as no use of emergency fans, calculation CA02725 shows a CR temperature of 123 deg F at 24-hours. While this is a challenging environment, this temperature is assessed as insufficient to solely drive a complete CR abandonment scenario. NUREG/CR-6738 describes operational experience where operators will continue to occupy the control room even under severe environments. Operations staff says that in consideration of high temperatures in the control room, that Operations would do what was needed to keep the cores safe and covered. The site safety director says that for a temperature of 123 deg F, the site would implement a mitigation strategy which would include stay-times, assessment of individuals for heat-related conditions, use of ice vests, and call-in of additional qualified operations staff to rotate into the control room. The above discussion was included in C0-SY-030, Control Room HVAC PRA System Notebook.

Table B-2 Fire PRA – Category I Summary

SR	Topic	Status
FSS-A5	2012 Peer Review: SR Not Met A range of ignition source / target set combinations has been represented for unscreened PAUs. These combinations are identified in relevant calculation sheets for unscreened PAUs (filenames RSC-CALKNX-2011-xxx.pdf). However, it is not clear how the potential for spreading fires and for fire and smoke spread between sub-PAUs is addressed and consequently it cannot be determined if the selected fire scenarios provide reasonable assurance that the risk contribution of each unscreened PAU can be characterized. Associated F&O: FSS-A5-01	Now: Met Cat I/II FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. In both cases, thermocouple location was adjusted as identified in F&O FSS-D3-02. For the CSR, consequences were divided into scenarios based on mitigation potential. First, if the scenario was suppressed by the Halon system then the limit of damage was based on what was predicted by FDS in terms of temperature and energy. If it was unsuppressed it went to total room burn, which assumes failure of all targets in the room, regardless of the initial scenario boundary. For the Switchgear Room FDS analysis, the analysis was updated to add clarity to the analysis. A discussion of the application of sub-PAUs has been added to Addendum 1 to C0-FSS-004, Fire PRA Detailed Fire Modeling Notebook. Damage was not limited to specified sub-PAUs. Specific examples of the treatment of fire growth and the application of sub-PAUs has been provided. As described in C0-FSS-004, the sub-PAU analysis included spatial information from walkdown, along with engineering judgment, to determine if fire sources could fail additional components, cables, or other combustibles, potentially leading to more damage to surrounding equipment or cables. For scenarios that leveraged FDT modeling, the issue related to whether the analysis had correctly addressed the impact of transients along the edge of a boundary interface for a sub-PAU. A comparable consideration was also related to secondary combustion and oil fires. Resolution involved selection of several representative PAUs for a sensitivity study that expanded the existing sub-PAUs and examined secondary ignition potential. The PAUs were considered representative of the work performed based on several criteria. The analysis indicated that the methods mentioned were indeed appropriate. Sub-PAU impacts did not change from the expanded assessment and that secondary ignition was bounded by the existing analysis and was appropriately addressed. The analysis was incorporated into the documentation for C0-FSS-004.

Table B-2 Fire PRA – Category I Summary

SR	Topic	Status
FSS-D3	2012 Peer Review: SR Not Met This SR is not met for multiple reasons. First, detailed FDS fire modeling evaluations of PAUs 302, 306, 311, 317, 407 and 430 assume that material surfaces are "inert." As noted on p. 44 of C0-FSS-004 R1, this assumption was made " so that no objects in the PAU or the PAU structure (walls, floor, or ceiling) itself would absorb any heat from the various fire scenarios, producing a more conservative or worst case result for all fire scenarios' impacts to the components and cables within the PAU model. As such, no detailed material properties were required to be defined in FDS for the scenarios to function correctly." However, specification of material surfaces as "inert" in FDS does not prevent heat absorption into material surfaces. On the contrary, this specification maintains material surfaces at ambient temperature in FDS, which tends to maximize heat absorption into these surfaces. To meet the specified goal of preventing heat absorption into material surfaces, they should have been specified as "adiabatic" rather than as "inert." (See Finding FSS-D3-01) Second, temperature measurement locations specified in the detailed FDS fire modeling evaluations do not generally coincide with locations where maximum temperatures are expected (e.g., within the fire plume). As a consequence, for some fire scenarios damage to targets is not predicted when it should be based on the specified damage criteria. (See Finding FSS-D3-02) Associated F&Os: FSS-D3-01 and FSS-D3-02	Now: Met Cat. II. FDS modeling was used for fire scenario evaluations in the Cable Spreading Rooms and Switchgear Rooms. <u>Material Properties</u> For the Cable Spreading Room FDS fire scenarios, the Unit 1 CSR was modified to include actual material properties and sensitivity analysis. Actual material properties were used in the updated U1CSR FDS model rather than the prior use of "inert" material conditions. Adiabatic conditions were used for any items with material properties that are unknown or of a high uncertainty to bound the analysis and prevent heat transfer into those objects. The CSR FDS model was executed and the results compared to the baseline results. This study was then documented in FSS-004. The results were applied to Unit 2 CSR. This study was then documented in FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' Switchgear Rooms were updated to specify representative material properties as referenced by NUREG 1805. This adjustment enabled the analysis to obtain more realistic estimates of environmental conditions for these fire scenarios. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1. <u>Temperature Measurement Locations</u> For the Cable Spreading Room FDS fire scenarios, new measurement devices were included in the updated U1CSR FDS model. The thermocouples were placed directly above the fire source in the updated FDS model and the scenarios re-evaluated. The results were applied to Unit 2 CSR. This study and the results were then documented in FSS-004, Fire PRA Detailed Fire Modeling Notebook. The Unit 1 27' and 45' SWGR rooms were updated to alter the location of the thermocouples such that the centerline plume temperature was recorded and used to determine target impacts. Results calculated in the Unit 1 FDS models were applied to Unit 2. Results of the updated model are incorporated into C0-FSS-004 as Addendum 1.

Table B-2 Fire PRA – Category I Summary

SR	Topic	Status
FSS-F3	2012 Peer Review: SR Met: (CC I) A number of potential scenarios are selected and a qualitative assessment of the associated risk is performed for the selected fire scenarios. Associated F&O: FSS-F3-01	Now: Met Cat. II/III This subject of this SR are fire-induced damage to structural steel. As described in C0-FSS-005, Calvert Cliffs Fire PRA Structural Failure Analysis Notebook, the un-screened structural steel scenarios are in the Turbine Building. The Turbine Building analysis was reviewed for potential fire scenarios where structural steel can be adversely affected. From the scenarios examined, those that can damage structural steel were selected for further analysis. The frequency, severity factor and non-suppression probability of each scenario were developed and included in the Structural Failure Analysis Notebook. These impacts were then added to FRANX database and quantified as part of the final Fire PRA risk quantification in Fire Quantification Notebooks C0-FRQ-001 and C0-FRQ-002.
FSS-G4	2012 Peer Review: SR Met: (CC I) Passive fire barriers are credited in the multi-compartment analysis consistent with fire resistance ratings, but the effectiveness, reliability and availability of credited passive fire barriers have not been assessed. Associated F&O: FSS-G4-01	Now: Met Cat. II Generic probabilities were used for credited passive fire barrier features in the multi-compartment analysis. At Calvert Cliffs, the fire barriers are verified to be effective through test procedures. An unreliability value was applied to all normally closed doors that represents the probability of the door being propped open given a fire in the exposing compartment. The probability of finding a failed sealed wall penetration is assumed to be very small to warrant propagation scenarios. A discussion of the effectiveness, reliability, and availability of fire barriers was added to C0-FSS-008, Calvert Fire PRA Multi-Compartment Analysis.
FSS-G5	2012 Peer Review: SR Met: (CC I) The effectiveness, reliability and availability of active fire barrier elements has been assessed qualitatively, but has not been quantified. Associated F&O: FSS-G5-01	Now: Met Cat. II/III Active fire barriers were evaluated as effective in studies used to support Appendix R analysis. An unreliability value has been applied to all normally open, self closing dampers and doors; A discussion of the effectiveness of credited active fire barriers was added to C0-FSS-008, Calvert Fire PRA Multi-Compartment Analysis.

C. Total CDF, LERF and RG 1.174

The PRA scope that is currently developed in accordance with the ASME/ANS RA-Sa-2009 standard and Reg. Guide 1.200 are fire and internal events. Table C-1 includes the values for the following:

- Fire: The CDF and LERF are obtained from the Fire PRA. These risk metrics take into account the proposed plant modifications credited for the NFPA 805 transition. Section B discusses PRA quality for the fire model.
- Internal events (including internal floods): CDF and LERF used to evaluate the total plant risk are based on preliminary estimates, taking into account the proposed plant modifications credited for the NFPA 805 transition. Section A discusses PRA quality for the internal events model.

The following external hazards were screened in the IPEEE as not significantly influencing total risk. There is no PRA model for these hazards:

- External Flooding,
- Transportation accidents, and
- Industrial Accidents.

Calvert Cliffs does not have a low power or shutdown PRA model.

Of the hazards evaluated in the Individual Plant Examination of External Events (IPEEE), tornadoes/high winds and seismic events were of note. Table C-1 uses the IPEEE CDF values for tornadoes/high winds and seismic events, and for these, Table C-1 uses an estimate of the LERF values based on a reasonable 13% fraction of CDF for the following:

- Seismic events: PRA risk for seismic events is based upon IPEEE values. Since the Calvert Seismic PRA has not been updated since the IPEEE, the results of the seismic PRA must be qualitatively evaluated for impact on the STI extension.

  The NRC evaluated seismic risk generically in ML100270756. As shown for the 2008 Seismic Hazard Curves in Table D-1 of ML100270756, the "weakest link" estimated CDF risk for Calvert Cliffs Unit 1 is 1.0E-05 and for Unit 2 is 1.2E-05. These updated CDF values from ML100270756 are bounded by the IPEEE results presented in Table C-1.

  The Calvert Cliffs SPRA was performed using the methodology outlined in NUREG-1407. The Seismic hazard curves used are those developed for CCNPP by Lawrence Livermore National Laboratory (LLNL) and provided in NUREG-1488. In late 2013 EPRI provided Calvert Cliffs a new Ground Motion Response Spectrum (GRMS). A preliminary comparison of the new Calvert Cliffs GMRS to the Calvert Cliffs IPEEE HCLPF Spectra (IHS) shows that the GMRS is lower than the IHS at all frequencies. This is true for GMRS comparisons utilizing both the mean and median hazards as inputs in the IHS development. Based on these comparisons, the new GMRS is not expected to have a significant impact on seismic CDF at Calvert Cliffs.
- Wind events: PRA risk for tornadoes and high winds are based upon IPEEE values. Calvert Cliffs has maintained and updated a high wind PRA model in order to perform risk assessment of tornado missile impacts and hurricane force winds. Although this model has not been peer reviewed in compliance with the ASME/ANS standard, the model is

based upon accepted methodology and utilizes the ASME/ANS compliant internal events model. A recent quantification of the wind initiating events using the updated internal events model estimates CDF risk for Calvert Cliffs Unit 1 at 9.4E-07 and for Unit 2 at 9.4E-07. These updated CDF values are bounded by the IPEEE results presented in Table C-1.

In addition, the tornado/high wind and the seismic results in Table C-1 do not credit risk reduction for the NFPA 805 modifications. This further conservatively bounds the summary of total plant risk values in the table.

Table C-1 - Summary of Total Plant Risk for Calvert Cliffs

Event Type	Unit 1 (/rx-yr)		Unit 2 (/rx-yr)		Comments
	CDF	LERF	CDF	LERF	
Fire	3.2E-05	3.2E-06	3.6E-05	4.4E-06	Fire PRA quantification with NFPA 805 modifications credited.
Internal Events (including internal floods)	1.3E-05	1.3E-06	1.0E-05	1.3E-06	Internal Events PRA quantification with NFPA 805 modifications credited.
Seismic Events	1.3E-05	1.7E-06	1.5E-05	2.0E-06	IPEEE for CDF. Estimate for LERF based on 13% of CDF.
Tornadoes/High Winds	4.4E-06	5.7E-07	4.4E-06	5.7E-07	IPEEE for CDF. Estimate for LERF based on 13% of CDF.
Plant-Level Total	**6.2E-05**	**6.8E-06**	**6.5E-05**	**8.3E-06**	

The data from Table C-1 is derived either from an ASME/ANS compliant PRA model or from reasonable or bounding PRA analysis for models that have not had an ASME/ANS peer review. There is high confidence that the overall CDF is less than the Regulatory Guide 1.174 limits for total plant risk of 1E-04 per year and LERF is less than 1E-05 per year

ATTACHMENT (3)

MARKED-UP TECHNICAL SPECIFICATION PAGES

Insert 1

In accordance with the Surveillance Frequency Control Program

Insert 2

5.5.19 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 Technical Specifications Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies," Revision 1.
c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

1.1 Definitions

	verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length control element assemblies (CEAs) (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. With any CEAs not capable of being fully inserted, the reactivity worth of these CEAs must be accounted for in the determination of SDM.
~~STAGGERED TEST BASIS~~	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM is within limits specified in the COLR.	~~24 hours~~ Insert 1
SR 3.1.1.2	-------------------NOTE------------------- Only required in MODE 5 with pressurizer level < 90 inches. -- Verify Reactor Coolant System level is above the bottom of the hot leg nozzles.	Once within 1 hour after achieving MODE 5 with pressurizer level < 90 inches <u>AND</u> 12 hours thereafter

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition C, D, or E not met. OR One or more CEAs untrippable. OR Two or more CEAs misaligned by > 15 inches.	F.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify the indicated position of each CEA to be within 7.5 inches of all other CEAs in its group.	Within 1 hour following any CEA movement of > 7.5 inches AND Insert 1 ~~12 hours~~
SR 3.1.4.2	Verify the CEA motion inhibit is OPERABLE.	~~31 days~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.1.4.3	Verify the CEA deviation circuit is OPERABLE.	~~31 days~~
SR 3.1.4.4	Verify CEA freedom of movement (trippability) by moving each individual CEA that is not fully inserted into the reactor core 7.5 inches in either direction.	~~92 days~~ Insert 1
SR 3.1.4.5	Perform a CHANNEL FUNCTIONAL TEST of the reed switch position transmitter channel.	~~24 months~~
SR 3.1.4.6	Verify each CEA drop time is ≤ 3.1 seconds.	Prior to reactor criticality, after each removal of the reactor head

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One shutdown CEA withdrawn ≥ 121.5 inches and < 129 inches for > 7 days per occurrence or > 14 days per 365 days. OR One shutdown CEA withdrawn < 121.5 inches. OR Two or more shutdown CEAs not within limit.	B.1 Restore shutdown CEA(s) to within limit.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown CEA is withdrawn ≥ 129 inches.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify each regulating CEA group position is within its insertion limits.	~~12 hours~~
SR 3.1.6.2	Verify the accumulated times during which the regulating CEA groups are inserted beyond the steady state insertion limits, but within the transient insertion limits.	~~24 hours~~ Insert 1
SR 3.1.6.3	Verify power dependent insertion limit alarm circuit is OPERABLE.	~~31 days~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (Continued) All CEAs inserted and the reactor subcritical by less than the above shutdown reactivity equivalent.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify that the position of each CEA not fully inserted is within the acceptance criteria for available negative reactivity addition.	~~2 hours~~ Insert 1
SR 3.1.7.2	-------------------NOTE------------------- Not required to be performed during initial power escalation following a refueling outage if SR 3.1.4.6 has been met. --- Verify that each CEA not fully inserted is capable of full insertion when tripped from at least the 50% withdrawn position.	Once within 7 days prior to reducing SDM to less than the limits of LCO 3.1.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Suspend PHYSICS TESTS. AND B.2 Be in MODE 3.	1 hour 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify THERMAL POWER is equal to or less than the test power plateau.	~~1 hour~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
Either the Excore Detector Monitoring System or the Incore Detector Monitoring System shall be used to determine LHR.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Deleted	
SR 3.2.1.2	-------------------NOTE------------------- Only applicable when the Excore Detector Monitoring System is being used to determine LHR. -- Verify ASI alarm setpoints are within the limits specified in the COLR.	Insert 1 ~~31 days~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	------------------ NOTES ------------------ 1. Only applicable when the Incore Detector Monitoring System is being used to determine LHR. 2. Not required to be performed below 20% RTP. --- Verify incore detector local power density alarms satisfy the requirements of the core power distribution map, which shall be updated at least once per 31 days of accumulated operation in MODE 1.	~~31 days~~
SR 3.2.1.4	------------------ NOTES ------------------ 1. Only applicable when the Incore Detector Monitoring System is being used to determine LHR. 2. Not required to be performed below 20% RTP. --- Verify incore detector local power density alarm setpoints are less than or equal to the limits specified in the COLR.	Insert 1 ~~31 days~~

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.3.1	-------------------NOTE------------------- F_r^T shall be determined by using the incore detectors to obtain a power distribution map with all full length control element assemblies at or above the long-term steady state insertion limit as specified in the COLR. --	
	Verify the value of F_r^T.	Prior to operation > 70% RTP after each fuel loading <u>AND</u> Insert 1 ~~Each 31 days of accumulated operation in MODE 1~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	Verify T_q is within limits.	~~12 hours~~ Insert 1

3.2 POWER DISTRIBUTION LIMITS

3.2.5 AXIAL SHAPE INDEX (ASI)

LCO 3.2.5 The ASI shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ASI not within limits.	A.1 Restore ASI to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 20% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.5.1 Verify ASI is within limits specified in the COLR.	~~12 hours~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time not met for Axial Power Distribution-High and Loss of Load Trip Functions.	F.1 Reduce THERMAL POWER to < 15% RTP.	6 hours
G. Required Action and associated Completion Time not met except for Axial Power Distribution-High and Loss of Load Trip Functions.	G.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which Surveillance Requirement shall be performed for each RPS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform a CHANNEL CHECK of each RPS instrument channel except Loss of Load.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.2	------------------ NOTES ------------------ 1. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. 2. The daily calibration may be suspended during PHYSICS TESTS, provided the calibration is performed upon reaching each major test power plateau, and prior to proceeding to the next major test power plateau. -- Perform a calibration (heat balance only) and adjust the excore power range and ΔT power channels to agree with calorimetric calculation if the absolute difference is ≥ 1.5%.	~~24 hours~~ Insert 1
SR 3.3.1.3	------------------ NOTE ------------------ Not required to be performed until 12 hours after THERMAL POWER is ≥ 20% RTP and required to be performed prior to operation above 90% RTP. -- Calibrate the power range excore channels using the incore detectors.	~~31 days~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.4	Perform a CHANNEL FUNCTIONAL TEST of each RPS instrument channel except Loss of Load and Rate of Change of Power-High.	~~92 days~~
SR 3.3.1.5	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform a CHANNEL CALIBRATION on excore power range channels.	Insert 1 ~~92 days~~
SR 3.3.1.6	Perform a CHANNEL FUNCTIONAL TEST of each Rate of Change of Power-High and Loss of Load instrument channel.	Once within 7 days prior to each reactor startup
SR 3.3.1.7	Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal feature.	~~24 months~~
SR 3.3.1.8	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform a CHANNEL CALIBRATION of each instrument channel, including applicable automatic bypass removal functions.	Insert 1 ~~24 months~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	-------------------NOTE------------------- Neutron detectors are excluded from RPS RESPONSE TIME testing. -- Verify RPS RESPONSE TIME is within limits.	Insert 1 ~~24 months on a STAGGERED TEST BASIS~~

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1	Perform a CHANNEL CHECK of each Wide Range Logarithmic Neutron Flux Monitor.	~~12 hours~~ Insert 1
SR 3.3.2.2	Perform a CHANNEL FUNCTIONAL TEST on the Rate of Change of Power trip instrument channel. The allowable value shall be ≤ 2.6 dpm.	Once within 7 days prior to each reactor startup
SR 3.3.2.3	Perform a CHANNEL FUNCTIONAL TEST on each automatic bypass removal feature.	~~24 months~~ Insert 1
SR 3.3.2.4	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform a CHANNEL CALIBRATION, including automatic bypass removal features.	~~24 months~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two channels of RTCBs or Trip Path Logic affecting the same trip leg inoperable.	D.1 Open the affected RTCBs.	Immediately
E. Required Action and associated Completion Time of Condition A, B, or D not met. OR One or more Functions with two or more Manual Trip, Matrix Logic, Trip Path Logic, or RTCB channels inoperable for reasons other than Condition A or D.	E.1 Be in MODE 3. AND E.2 Open all RTCBs.	6 hours 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform a CHANNEL FUNCTIONAL TEST on each RTCB channel.	~~92 days~~ Insert 1
SR 3.3.3.2	Perform a CHANNEL FUNCTIONAL TEST on each RPS Logic channel.	~~92 days~~

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.4.1	Perform a CHANNEL CHECK of each ESFAS sensor channel.	~~12 hours~~
SR 3.3.4.2	Perform a CHANNEL FUNCTIONAL TEST of each ESFAS sensor channel.	~~92 days~~
SR 3.3.4.3	Perform a CHANNEL FUNCTIONAL TEST on each automatic block removal feature.	~~24 months~~ Insert 1
SR 3.3.4.4	Perform a CHANNEL CALIBRATION of each ESFAS sensor channel, including automatic block removal feature.	~~24 months~~
SR 3.3.4.5	Verify ESF RESPONSE TIME is within limits.	~~24 months on a STAGGERED TEST BASIS~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	------------------ NOTES ------------------ 1. Testing of Actuation Logic shall include verification of the proper relay driver output signal. 2. Relays associated with plant equipment that cannot be operated during plant operation are only required to be tested once per 24 months. -- Perform a CHANNEL FUNCTIONAL TEST on each ESFAS Actuation Logic channel.	~~92 days~~ Insert 1
SR 3.3.5.2	Perform a CHANNEL FUNCTIONAL TEST on each ESFAS Manual Actuation channel.	~~24 months~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Enter applicable Conditions and Required Actions for the associated DG made inoperable by DG-LOVS instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL FUNCTIONAL TEST.	~~92 days~~ Insert 1
SR 3.3.6.2	Perform CHANNEL CALIBRATION with setpoint Allowable Values as follows: 1. Transient Degraded Voltage Function $\geq$ 3630 V and $\leq$ 3790 V; Time Delay: $\geq$ 7.6 seconds and $\leq$ 8.4 seconds; 2. Steady State Degraded Voltage Function $\geq$ 3820 V and $\leq$ 3980 V Time Delay: $\geq$ 97.5 seconds and $\leq$ 104.5 seconds; and 3. Loss of voltage Function $\geq$ 2345 V and $\leq$ 2555 V Time Delay: $\geq$ 1.8 seconds and $\leq$ 2.2 seconds at 2450 V.	~~24 months~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required Manual Actuation channel or Actuation Logic channel inoperable. OR More than one radiation monitor sensor module or associated measurement channel inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1 Place and maintain containment purge and exhaust valves in closed position. OR B.2 Enter applicable Conditions and Required Actions for affected valves of LCO 3.9.3, "Containment Penetrations," made inoperable by isolation instrumentation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform a CHANNEL CHECK on each containment radiation monitor sensor.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.7.2	-------------------NOTE------------------- Testing of Actuation Logic shall include verification of the proper relay driver output signal. -- Perform a CHANNEL FUNCTIONAL TEST on each CRS Actuation Logic channel.	~~92 days~~
SR 3.3.7.3	Perform a CHANNEL FUNCTIONAL TEST on each containment radiation monitor sensor. Verify CRS high radiation setpoint is less than or equal to the Allowable Value of 220 mR/hr.	~~92 days~~ Insert 1
SR 3.3.7.4	Perform a CHANNEL CALIBRATION on each containment radiation monitor instrument channel.	~~24 months~~
SR 3.3.7.5	Perform a CHANNEL FUNCTIONAL TEST on each CRS Manual Actuation channel.	~~24 months~~
SR 3.3.7.6	Verify CRS response time is within limits.	~~24 months on a STAGGERED TEST BASIS~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. CRRS trip circuit or measurement channel inoperable during movement of irradiated fuel assemblies.	C.1 Place one Control Room Emergency Ventilation System train in recirculation mode with post-loss-of-coolant incident filter fan in service.	Immediately
	<u>OR</u>	
	C.2 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Perform a CHANNEL CHECK on the control room radiation monitor channel.	~~12 hours~~ Insert 1
SR 3.3.8.2	Perform a CHANNEL FUNCTIONAL TEST on the CRRS radiation monitor trip circuit and measurement channel. Verify CRRS high radiation setpoint is less than or equal to the Allowable Value of 6E4 cpm above normal background.	~~92 days~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.8.3	Perform a CHANNEL CALIBRATION on the CRRS radiation monitor trip circuit and measurement channel.	~~24 months~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two CVCS isolation sensor modules or associated measurement channels inoperable.	C.1	Place one sensor module in bypass and place the other sensor module in trip.	1 hour
	AND		
	C.2	Restore one sensor module and associated measurement channel to OPERABLE status.	48 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform a CHANNEL CHECK of each sensor channel.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.9.2	------------------ NOTES ------------------ 1. Testing of Actuation Logic shall include the verification of the proper relay driver output signal. 2. Relays associated with plant equipment that cannot be operated during plant operation are only required to be tested once per 24 months. --- Perform a CHANNEL FUNCTIONAL TEST on each CVCS sensor channel with setpoints in accordance with the following Allowable Values: West Penetration Room Pressure-High ≤ 0.5 psig Letdown Heat Exchanger Room Pressure-High ≤ 0.5 psig	~~92 days~~ Insert 1
SR 3.3.9.3	Perform a CHANNEL CALIBRATION on each CVCS sensor channel.	~~24 months~~
SR 3.3.9.4	Verify CVCS Isolation Signal response time is within limits.	~~24 months on a STAGGERED TEST BASIS~~

SURVEILLANCE REQUIREMENTS

-----NOTE-----
These Surveillance Requirements apply to each PAM instrumentation Function in Table 3.3.10-1.

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	Perform CHANNEL CHECK for each required indication channel that is normally energized.	~~31 days~~
SR 3.3.10.2	Deleted	Insert 1
SR 3.3.10.3	-----NOTE----- Neutron detectors, Core Exit Thermocouples, and Reactor Vessel Level Monitoring System are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION on each indication channel.	~~24 months~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.11.1	Perform CHANNEL CHECK for each required indication channel that is normally energized.	~~31 days~~
SR 3.3.11.2	-------------------NOTE------------------- Neutron detectors and Reactor Trip Breaker Indication are excluded from the CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION for each required indication channel.	Insert 1 ~~24 months~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.12.1	Perform CHANNEL CHECK.	~~12 hours~~
SR 3.3.12.2	Perform CHANNEL FUNCTIONAL TEST.	Once within 7 days prior to each reactor startup
SR 3.3.12.3	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	Insert 1 ~~24 months~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within the limits specified in the COLR.	~~12 hours~~
SR 3.4.1.2	Verify RCS cold leg temperature is within the limits specified in the COLR.	~~12 hours~~ Insert 1
SR 3.4.1.3	Verify RCS total flow rate is greater than or equal to the limits specified in the COLR.	~~12 hours~~
SR 3.4.1.4	Verify measured RCS total flow rate is within the limits specified in the COLR.	~~24 months~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (Continued)	B.2 Be in MODE 5 with RCS pressure < 300 psia.	36 hours
C. ---------NOTE------- Required Action C.2 shall be completed whenever this Condition is entered. --------------------- Requirements of Limiting Condition for Operation not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
	C.2 Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within limits specified in Figures 3.4.3-1 and 3.4.3-2.	Insert 1 ~~30 minutes~~

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODES 1 and 2

LCO 3.4.4 Two RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of Limiting Condition of Operation not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify each RCS loop is in operation.	~~12 hours~~ Insert 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours
C. No RCS loop OPERABLE. <u>OR</u> No RCS loop in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1. <u>AND</u> C.2 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.5.1 Verify required RCS loop is in operation.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.5.2	Verify secondary side water level in each steam generator > -50 inches.	~~12 hours~~ Insert 1
SR 3.4.5.3	Verify correct breaker alignment and indicated power available to the required pump that is not in operation.	~~7 days~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RCS or SDC loop is in operation.	~~12 hours~~
SR 3.4.6.2	Verify secondary side water level in required steam generator(s) is > -50 inches.	~~12 hours~~ Insert 1
SR 3.4.6.3	Verify correct breaker alignment and indicated power available to the required loop components that are not in operation.	~~7 days~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one SDC loop is in operation.	~~12 hours~~
SR 3.4.7.2	Verify required SG secondary side water level is > -50 inches.	~~12 hours~~
SR 3.4.7.3	Verify correct breaker alignment and indicated power available to the required SDC loop components that are not in operation.	~~7 days~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required SDC loops inoperable. OR No SDC loop in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1.	Immediately
	AND	
	B.2 Initiate action to restore one SDC loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify one SDC loop is in operation.	~~12 hours~~
SR 3.4.8.2	Verify correct breaker alignment and indicated power available to the required SDC loop components that are not in operation.	~~7 days~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. AND C.2 Be in Mode 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≥ 133 inches and ≤ 225 inches.	~~12 hours~~ Insert 1
SR 3.4.9.2	Verify capacity of each required bank of pressurizer heaters ≥ 150 kW.	~~24 months~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two block valves inoperable.	E.1 Place associated PORVs in override closed. <u>AND</u> E.2 Restore one block valve to OPERABLE status.	1 hour 72 hours
F. Required Action and associated Completion Time not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2).	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Perform a CHANNEL FUNCTIONAL TEST of each PORV.	~~92 days~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.11.2	-------------------NOTE------------------- Not required to be performed with block valve closed in accordance with the Required Actions of this Limiting Condition for Operation. -- Perform a complete cycle of each block valve.	~~92 days~~ Insert 1
SR 3.4.11.3	Perform a complete cycle of each PORV.	~~24 months~~
SR 3.4.11.4	Perform a CHANNEL CALIBRATION of each PORV.	~~24 months~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of one HPSI pump is only capable of manually injecting into the RCS.	~~12 hours~~
SR 3.4.12.2	Verify HPSI loop MOVs are only capable of manually aligning HPSI pump flow to the RCS.	~~12 hours~~
SR 3.4.12.3	Verify required RCS vent is open.	~~12 hours for unlocked open vent valve(s)~~ ~~31 days for locked open vent valve(s)~~
SR 3.4.12.4	Verify PORV block valve is open for each required PORV.	~~72 hours~~
SR 3.4.12.5	-------------------NOTE------------------- Not required to be performed until 12 hours after decreasing any RCS cold leg temperature to ≤ 365°F (Unit 1), ≤ 301°F (Unit 2). -- Perform CHANNEL FUNCTIONAL TEST on each required PORV, excluding actuation.	Insert 1 ~~31 days~~
SR 3.4.12.6	Perform CHANNEL CALIBRATION on each required PORV actuation channel.	~~24 months~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	------------------NOTES------------------ 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS Operational LEAKAGE is within limits by performance of RCS water inventory balance.	Insert 1 ~~72 hours~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.13.2	-------------------NOTE------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is ≤ 100 gallons per day through any one SG.	Insert 1 ~~72 hours~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
E. All required alarms and monitors inoperable.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitor.	~~12 hours~~
SR 3.4.14.2	Perform CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitor.	~~31 days~~ Insert 1
SR 3.4.14.3	Perform CHANNEL CALIBRATION of the required containment sump level alarm.	~~24 months.~~
SR 3.4.14.4	Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor.	~~24 months.~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.15-1.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours
C. Gross activity of the reactor coolant not within limit.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Verify reactor coolant gross activity $\leq 100/\bar{E}$ μCi/gm.	~~7 days~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.15.2	-------------------NOTE------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.5 μCi/gm.	Insert 1 ~~14 days~~ AND Between 2 and 6 hours after THERMAL POWER change of ≥ 15% RTP within a 1 hour period
SR 3.4.15.3	-------------------NOTE------------------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. --- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	Insert 1 ~~184 days~~

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 Special Test Exception (STE) RCS Loops - MODE 2

LCO 3.4.16 The requirements of LCO 3.4.4, "RCS Loops-MODES 1 and 2," and the listed requirements of LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation-Operating," for the Reactor Coolant Flow-Low, Thermal Margin/Low Pressure, and Asymmetric Steam Generator Transient Functions may be suspended provided:

a. THERMAL POWER ≤ 5% RTP; and

b. The reactor trip setpoints of the OPERABLE Power Level-High channels are set ≤ 15% RTP.

APPLICABILITY: MODE 2, during startup and PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Open reactor trip breakers.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify THERMAL POWER ≤ 5% RTP.	~~1 hour~~ Insert 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more requirements of the Limiting Condition for Operation not met.	A.1 Suspend activities being performed under this Special Test Exception.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify xenon reactivity is within limits.	Once within 1 hour prior to suspending the reactor coolant circulation requirements of LCO 3.4.6, LCO 3.4.7, and LCO 3.4.8
SR 3.4.17.2	Verify charging pumps de-energized.	~~1 hour~~
SR 3.4.17.3	Verify charging flow paths isolated.	~~1 hour~~ Insert 1
SR 3.4.17.4	Perform SR 3.1.1.1.	~~8 hours~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each SIT isolation valve is fully open.	~~12 hours~~
SR 3.5.1.2	Verify borated water volume in each SIT is ≥ 1113 cubic feet (187 inches) and ≤ 1179 cubic feet (199 inches).	~~12 hours~~ Insert 1
SR 3.5.1.3	Verify nitrogen cover pressure in each SIT is ≥ 200 psig and ≤ 250 psig.	~~12 hours~~
SR 3.5.1.4	Verify boron concentration in each SIT is ≥ 2300 ppm and ≤ 2700 ppm.	~~12 hours (by inleakage monitoring)~~ ~~AND~~ ~~6 months (by sample)~~ AND -----NOTE ---- Only required to be performed for affected SIT -------------- Once within 1 hour prior to each solution volume increase of ≥ 1% of tank volume

SURVEILLANCE		FREQUENCY
SR 3.5.1.5	Verify power is removed from each SIT isolation valve operator when pressurizer pressure is ≥ 2000 psig.	~~31 days~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed. Valve Number / Position / Function MOV-659 / Open / Mini-flow Isolation MOV-660 / Open / Mini-flow Isolation CV-306 / Open / Low Pressure Safety Injection Flow Control	~~12 hours~~ Insert 1
SR 3.5.2.2	Verify each ECCS manual, power-operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~
SR 3.5.2.3	Verify each high pressure safety injection and low pressure safety injection pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.4	Deleted	
SR 3.5.2.5	Verify each ECCS automatic valve that is not locked, sealed, or otherwise secured in position, in the flow path actuates to the correct position on an actual or simulated actuation signal.	~~24 months~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	~~24 months~~
SR 3.5.2.7	Verify each low pressure safety injection pump stops on an actual or simulated actuation signal.	~~24 months~~
SR 3.5.2.8	Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.	~~24 months~~ Insert 1
SR 3.5.2.9	Verify the Shutdown Cooling System open-permissive interlock prevents the Shutdown Cooling System suction isolation valves from being opened with a simulated or actual Reactor Coolant System pressure signal of ≥ 309 psia.	~~24 months~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	------------------NOTE------------------ Only required to be performed when ambient air temperature is < 40°F. -- Verify RWT borated water temperature is ≥ 40°F.	~~24 hours~~
SR 3.5.4.2	------------------NOTES------------------ 1. Only required to be met in MODE 1. 2. Only required to be performed when ambient air temperature is > 100°F. -- Verify RWT borated water temperature is ≤ 100°F.	Insert 1 ~~24 hours~~
SR 3.5.4.3	Verify RWT borated water volume is ≥ 400,000 gallons.	~~7 days~~
SR 3.5.4.4	Verify RWT boron concentration is ≥ 2300 ppm and ≤ 2700 ppm.	~~7 days~~

3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

3.5.5 Sodium Tetraborate (STB)

LCO 3.5.5 The STB baskets shall contain ≥ 13,750 lbm of STB.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. STB not within limits.	A.1 Restore STB to within limits.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	Verify the STB baskets contain ≥ 13,750 lbm of equivalent weight sodium tetraborate decahydrate.	~~24 months~~ Insert 1
SR 3.5.5.2	Verify that a sample from the STB baskets provides adequate pH adjustment of water borated to be representative of a post-loss-of-coolant accident sump condition.	~~24 months~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	------------------ NOTES ------------------ 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. --- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	~~24 months~~

Insert 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.3.1	Verify each 4 inch containment vent valve is closed except when the 4 inch containment vent valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	~~31 days~~
SR 3.6.3.2	-------------------NOTE------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. -- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Insert 1 ~~31 days~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.3	-------------------NOTE------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. -- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic power-operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.5	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	~~24 months~~ Insert 1

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -1.0 psig and ≤ 1.0 psig.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limits.	~~12 hours~~ Insert 1

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 120°F.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.5.1 Verify containment average air temperature is within limit.	~~24 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power-operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	~~31 days~~ Insert 1
SR 3.6.6.2	Operate each containment cooling train fan unit for ≥ 15 minutes.	~~31 days~~
SR 3.6.6.3	Verify each containment cooling train cooling water flow rate is ≥ 2000 gpm to each fan cooler.	~~31 days~~
SR 3.6.6.4	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.5	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~24 months~~ Insert 1
SR 3.6.6.6	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	~~24 months~~
SR 3.6.6.7	Verify each containment cooling train starts automatically on an actual or simulated actuation signal.	~~24 months~~

3.6 CONTAINMENT SYSTEMS

3.6.8 Iodine Removal System (IRS)

LCO 3.6.8 Three IRS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One IRS train inoperable.	A.1 Restore IRS train to OPERABLE status.	7 days
B. Two IRS trains inoperable.	B.1 Restore one IRS train to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.8.1 Operate each IRS train for ≥ 15 minutes.	~~31 days~~

Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.8.2	Perform required IRS filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.6.8.3	Verify each IRS train actuates on an actual or simulated actuation signal.	~~24 months~~ Insert 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.3.1	Verify each AFW manual, power-operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine-driven pumps, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~ Insert 1
SR 3.7.3.2	Cycle each testable, remote-operated valve that is not in its operating position.	In accordance with the Inservice Testing Program
SR 3.7.3.3	-------------------NOTE------------------- Not required to be performed for the turbine-driven AFW pump until 24 hours after reaching 800 psig in the steam generators. -- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.3.4	-------------------NOTE------------------- Not required to be performed for the turbine-driven AFW pump until 24 hours after reaching 800 psig in the steam generators. -- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~24 months~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.3.5	-------------------NOTE------------------- Not required to be performed for the turbine-driven AFW pump until 24 hours after reaching 800 psig in the steam generators. -- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	~~24 months~~
SR 3.7.3.6	-------------------NOTE------------------- Not required to be performed for the AFW train with the turbine-driven AFW pump until 24 hours after reaching 800 psig in the steam generators. -- Verify the AFW system is capable of providing a minimum of 300 gpm nominal flow to each flow leg.	Insert 1 ~~24 months~~
SR 3.7.3.7	Verify the proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5 or 6 for > 30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify CST usable volume is ≥ 150,000 gallons per Unit.	~~12 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.1	-------------------NOTE------------------- Isolation of CC flow to individual components does not render the CC System inoperable. -- Verify each CC manual, power-operated, and automatic valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~ Insert 1
SR 3.7.5.2	Verify each CC automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~24 months~~
SR 3.7.5.3	Verify each CC pump starts automatically on an actual or simulated actuation signal.	~~24 months~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One SRW subsystem inoperable.	B.1	--------NOTE------- Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources--Operating," for diesel generator made inoperable by SRW. -------------------- Restore SRW subsystem to OPERABLE status.	 72 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 AND C.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	------------------NOTE------------------ Isolation of SRW flow to individual components does not render SRW inoperable. -- Verify each SRW manual, power-operated, and automatic valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.6.2	Verify each SRW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~24 months~~ Insert 1
SR 3.7.6.3	Verify each SRW pump starts automatically on an actual or simulated actuation signal.	~~24 months~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	------------------NOTE------------------ Isolation of SW System flow to individual components does not render SW inoperable. -- Verify each SW System manual, power-operated, and automatic valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~ Insert 1
SR 3.7.7.2	Verify each SW System automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~24 months~~
SR 3.7.7.3	Verify each SW System pump starts automatically on an actual or simulated actuation signal.	~~24 months~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Operate each required CREVS filter train for ≥ 15 minutes.	~~31 days~~ Insert 1
SR 3.7.8.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.8.3	Verify each CREVS train actuates on an actual or simulated actuation signal.	~~24 months~~
SR 3.7.8.4	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify each required CRETS train has the capability to maintain control room temperature within limits.	~~24 months~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify an OPERABLE SFPEVS train is in operation.	~~12 hours~~
SR 3.7.11.2	Deleted.	Insert 1
SR 3.7.11.3	Verify each SFPEVS fan can maintain a measurable negative pressure with respect to adjacent areas.	~~24 months~~

3.7 PLANT SYSTEMS

3.7.12 Penetration Room Exhaust Ventilation System (PREVS)

LCO 3.7.12 Two PREVS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PREVS train inoperable.	A.1 Restore PREVS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each PREVS train for ≥ 15 minutes.	~~31 days~~ Insert 1
SR 3.7.12.2	Verify required PREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.12.3	Verify each PREVS train actuates on an actual or simulated actuation signal.	~~24 months~~ Insert 1

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool (SFP) Water Level

LCO 3.7.13 The SFP water level shall be ≥ 21.5 ft over the top of irradiated fuel assemblies seated in the storage racks, and ≥ 19.8 ft over the top of fuel assemblies seated on rack spacers in the storage racks for reconstitution activities.

APPLICABILITY: During movement of irradiated fuel assemblies in the SFP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP water level not within limits.	A.1 ---------NOTE------- LCO 3.0.3 is not applicable. -------------------- Suspend movement of irradiated fuel assemblies in SFP and suspend reconstitution activities.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify the SFP water level is ≥ 21.5 ft above the top of irradiated fuel assemblies seated in the storage racks.	~~7 days~~ Insert 1

3.7 PLANT SYSTEMS

3.7.14 Secondary Specific Activity

LCO 3.7.14 The specific activity of the secondary coolant shall be $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the specific activity of the secondary coolant is within limit.	~~31 days~~

Insert 1

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Pool (SFP) Boron Concentration

LCO 3.7.16 Boron concentration of the SFP shall be ≥ 2000 ppm.

APPLICABILITY: When fuel assemblies are stored in the SFPs.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent Fuel Pool boron concentration not within limit.	----------- NOTE ------------ LCO 3.0.3 is not applicable. ---------------------------	
	A.1 Suspend movement of fuel assemblies in the SFPs.	Immediately
	<u>AND</u>	
	A.2 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify boron concentration is greater than 2000 ppm.	~~7 days~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. Three or more required LCO 3.8.1.a and LCO 3.8.1.b AC sources inoperable.	K.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
SR 3.8.1.1 through SR 3.8.1.15 are only applicable to LCO 3.8.1.a and LCO 3.8.1.b AC sources. SR 3.8.1.16 is only applicable to LCO 3.8.1.c AC sources.

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	-----NOTE----- Only required to be performed when SMECO is being credited for an offsite source. ----- Verify correct breaker alignment and indicated power availability for the 69 kV SMECO offsite circuit.	Once within 1 hour after substitution for a 500 kV offsite circuit AND ~~8 hours thereafter~~
SR 3.8.1.2	Verify correct breaker alignment and indicated power availability for each required 500 kV offsite circuit.	~~7 days~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.3	------------------NOTES------------------ 1. Performance of SR 3.8.1.9 satisfies this Surveillance Requirement. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this Surveillance Requirement as recommended by the manufacturer. When modified start procedures are not used, the voltage and frequency tolerances of SR 3.8.1.9 must be met. ---	
	Verify each DG starts and achieves steady state voltage ≥ 4060 V and ≤ 4400 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	~~31 days~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.4	------------------NOTES------------------ 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients below the load limit do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This Surveillance Requirement shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.3 or SR 3.8.1.9. --- Verify each DG is synchronized and loaded, and operates for ≥ 60 minutes at a load ≥ 4000 kW for DG 1A and ≥ 2700 kW for DGs 1B, 2A, and 2B.	~~31 days~~
SR 3.8.1.5	Verify each day tank contains ≥ 325 gallons of fuel oil for DG 1A and ≥ 275 gallons of fuel oil for DGs 1B, 2A, and 2B.	~~31 days~~ Insert 1
SR 3.8.1.6	Check for and remove accumulated water from each day tank.	~~31 days~~
SR 3.8.1.7	Verify the fuel oil transfer system operates to automatically transfer fuel oil from storage tank[s] to the day tank.	~~31 days~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.8	Verify interval between each sequenced load block is within ± 10% of design interval for the load sequencer.	~~31 days~~
SR 3.8.1.9	-------------------NOTE------------------- All DG starts may be preceded by an engine prelube period. --- Verify each DG starts from standby condition and achieves, in ≤ 10 seconds, voltage > 4060 V and frequency > 58.8 Hz, and after steady state conditions are reached, maintains voltage ≥ 4060 V and ≤ 4400 V and frequency of > 58.8 Hz and ≤ 61.2 Hz.	Insert 1 ~~184 days~~
SR 3.8.1.10	Verify manual transfer of AC power sources from the normal offsite circuit to the alternate offsite circuit.	~~24 months~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.11	------------------NOTE------------------ 1. Momentary transients outside the load and power factor limits do not invalidate this test. 2. If performed with the DG synchronized with offsite power, the surveillance test shall be performed at the required power factor. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition, the power factor shall be maintained as close to the limit as practicable. -- Verify each DG, operating at a power factor of ≤ 0.84 for DG 1A and ≤ 0.83 for DGs 1B, 2A, and 2B, operates for ≥ 60 minutes while loaded to ≥ 4000 kW for DG 1A and ≥ 3000 kW for DGs 1B, 2A, and 2B.	~~24 months~~ Insert 1
SR 3.8.1.12	Verify each DG rejects a load ≥ 500 hp without tripping.	~~24 months~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	Verify that automatically bypassed DG trips are automatically bypassed on an actual or simulated required actuation signal.	~~24 months~~ Insert 1
SR 3.8.1.14	Verify each DG: a. Synchronizes with offsite power source while loaded upon a simulated restoration of offsite power; b. Manually transfers loads to offsite power source; and c. Returns to ready-to-load operation.	~~24 months~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	-------------------NOTE------------------- All DG starts may be preceded by an engine prelube period. -- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated Engineered Safety Feature actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected emergency loads through load sequencer, 3. maintains steady state voltage ≥ 4060 V and ≤ 4400 V, 4. maintains steady state frequency of ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	~~24 months~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One or more DGs with new fuel oil properties not within limits.	E.1 Restore stored fuel oil properties to within limits.	30 days
F. Required Action and associated Completion Time not met. OR One or more DGs with diesel fuel oil not within limits for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated DG(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify fuel oil volume of: a. FOST 1A ≥ 49,500 gallons, and b. FOST 21 ≥ 85,000 gallons.	~~31 days~~ Insert 1
SR 3.8.3.2	Verify 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.	In accordance with the Diesel Fuel Oil Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.3	Check for and remove accumulated water from each FOST.	~~92 days~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 125 V on float charge.	~~7 days~~
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. OR Verify battery connection resistance is within limits.	~~92 days~~
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that degrades performance.	~~18 months~~ Insert 1
SR 3.8.4.4	Remove visible terminal corrosion and verify battery cell to cell and terminal connections are coated with anti-corrosion material.	~~18 months~~
SR 3.8.4.5	Verify battery connection resistance is within limits.	~~18 months~~
SR 3.8.4.6	Verify each battery charger supplies ≥ 400 amps at ≥ 125 V for ≥ 30 minutes.	~~24 months~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.7	-------------------NOTE------------------- The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7. -- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	~~24 months~~ Insert 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.8	Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	~~60 months~~ Insert 1 AND 12 months when battery shows degradation or has reached 85% of the expected life with capacity < 100% of manufacturer's rating AND 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more batteries with average electrolyte temperature of the representative cells < 69°F. OR One or more batteries with one or more battery cell parameters not within Category C limits.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	~~7 days~~ Insert 1
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	~~92 days~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ 69°F.	~~92 days~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage and alignment to required AC vital buses.	~~7 days~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.2	Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
		AND	
	A.2.3	Initiate action to restore required inverters to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct inverter voltage and alignment to required AC vital buses.	~~7 days~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.9.1 Verify correct breaker alignments and voltage to AC, DC, and AC vital bus electrical power distribution subsystems.	~~7 days~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.2	Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. AND	Immediately
	A.2.3	Initiate actions to restore required AC, DC, and AC vital bus electrical power distribution subsystems to OPERABLE status. AND	Immediately
	A.2.4	Declare associated required shutdown cooling subsystem(s) inoperable and not in operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems.	~~7 days~~ Insert 1

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System and the refueling pool shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend positive reactivity additions.	Immediately
	AND	
	A.2 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	~~72 hours~~ Insert 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	~~12 hours~~ Insert 1
SR 3.9.2.2	------------------NOTE------------------ Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	~~24 months~~

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	~~7 days~~ Insert 1
SR 3.9.3.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	~~24 months~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (Continued)	A.5 Close one door in each air lock.	4 hours
	AND	
	A.6.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
	OR	
	A.6.2 Verify each penetration is capable of being closed by an OPERABLE Containment Purge Valve Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one SDC loop is in operation and circulating reactor coolant at a flow rate of ≥ 1500 gpm.	~~12 hours~~ Insert 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (Continued)	B.5.2 Verify each penetration is capable of being closed by an OPERABLE Containment Purge Valve Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify required SDC loops are OPERABLE and one SDC loop is in operation.	~~12 hours~~
SR 3.9.5.2	Verify SDC loop in operation is circulating reactor coolant at a flow rate of ≥ 1500 gpm.	~~12 hours~~ Insert 1
SR 3.9.5.3	Verify correct breaker alignment and indicated power available to the required SDC loop components that are not in operation.	~~7 days~~

3.9 REFUELING OPERATIONS

3.9.6 Refueling Pool Water Level

LCO 3.9.6 Refueling pool water level shall be maintained ≥ 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling pool water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel.	~~24 hours~~ Insert 1

5.5 Programs and Manuals

Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. License controlled programs will be used to verify the integrity of the CRE boundary. Conditions that generate relevant information from those programs will be entered into the corrective action process and shall be trended and used as part of the 36 month assessments of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage 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 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and assessing the CRE boundary as required by paragraphs c and d respectively.

Insert 2

ATTACHMENT (4)

MARKED-UP TECHNICAL SPECIFICATION BASES PAGES

Insert 3

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

operator should borate with the best source available for the plant conditions. However, as a minimum, the boration flow rate shall be ≥ 40 gpm and the boron concentration shall be ≥ 2300 ppm boric acid solution or equivalent.

Assuming that a value of 1% Δk/k must be recovered and a boration flow rate of 40 gpm from the boric acid storage tank, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 15 minutes. If an inverse boron worth of 100 ppm/% Δk/k is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of 40 gpm and 100 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SHUTDOWN MARGIN is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. CEA positions;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient.

Using the isothermal temperature coefficient accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS.

Insert 3

~~The Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.~~

BASES

Required Action D.2.2 is modified by a Note indicating that performing this Required Action is not required when in conflict with Required Actions A.1, B.1, C.2, or E.1.

E.1

When the CEA deviation circuit is inoperable, performing SR 3.1.4.1 within 1 hour and every 4 hours thereafter ensures improper CEA alignments are identified before unacceptable flux distributions occur. The specified Completion Times take into account other information continuously available to the operator in the Control Room, so that during CEA movement, deviations can be detected, and the protection provided by the CEA inhibit and deviation circuit is not required.

F.1

If any Required Action and associated Completion Time of Condition C, Condition D, or Condition E is not met, one or more regulating or shutdown CEAs are untrippable, two or more CEAs are misaligned by > 15 inches, the unit is required to be brought to MODE 3. By being brought to MODE 3, the unit is brought outside the MODE of applicability. Continued operation is not allowed in the case of more than one CEA misaligned from any other CEA in its group by > 15 inches, or one or more CEAs untrippable. This is because these cases could result in a loss of SDM and power distribution and a loss of safety function, respectively.

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification is required ~~that~~ individual CEA positions are within 7.5 inches (indicated reed switch positions) of all other CEAs in the group ~~are performed at Frequencies of~~ within 1 hour of any CEA movement of > 7.5 inches ~~and every~~

~~12 hours~~. The CEA position verification after each movement of > 7.5 inches ensure that the CEAs in that group are properly aligned at the time when CEA misalignments are most likely to have occurred. ~~The 12 hour Frequency allows the operator to detect a CEA that is beginning to deviate from its expected position. The specified Frequency takes into account other CEA position information that is continuously available to the operator in the Control Room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA motion inhibit and deviation circuits~~.

Insert 3

SR 3.1.4.2

Demonstrating the CEA motion inhibit OPERABLE verifies that the CEA motion inhibit is functional, even if it is not regularly operated. The verification shall ensure that the motion inhibit circuit maintains the CEA group overlap and sequencing requirements of LCO 3.1.6, and prevents any regulating CEA from being misaligned from all other CEAs in its group by > 7.5 inches (indicated position). ~~The 31-day Frequency takes into account other information continuously available to the operator in the Control Room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA deviation circuits.~~

Insert 3

SR 3.1.4.3

Demonstrating the CEA deviation circuit is OPERABLE verifies the circuit is functional. ~~The 31-day Frequency takes into account other information continuously available to the operator in the Control Room, so that during CEA movement, deviations can be detected, and protection can be provided by the CEA motion inhibit.~~

Insert 3

SR 3.1.4.4

Verifying each CEA is trippable would require that each CEA be tripped. In MODEs 1 and 2, tripping each CEA would result in radial or axial power tilts or oscillations. Therefore, individual CEAs are exercised ~~every 92 days~~ to provide increased confidence that all CEAs continue to be trippable, even if they are not regularly tripped. A movement of 7.5 inches is adequate to demonstrate motion without exceeding the alignment limit when only one CEA is being moved. For the purposes of performing the CEA operability test, if the CEA has an inoperable position indicator channel, the alternate indication system (pulse counter or voltage dividing network) will be used to monitor position. ~~The 92-day Frequency takes into consideration other information available to the operator in the Control Room and other SRs being performed more frequently, which add to the determination of OPERABILITY of the CEAs~~.

Insert 3

Between required performances of SR 3.1.4.5, if a CEA(s)is discovered to be immovable, but remains trippable and aligned, the CEA is considered to be OPERABLE. At any time, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of the CEA(s) must be made, and appropriate action taken.

SR 3.1.4.5

Performance of a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position over the entire length of the CEA's travel. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. ~~Since this SR must be performed when the reactor is shut down, a 24-month Frequency to be coincident with refueling outages was selected. Operating experience has shown that these components usually pass this SR when performed at a Frequency of once every 24 months. Furthermore, the Frequency takes into account other SRs being performed at shorter Frequencies, which determine the OPERABILITY of the CEA Reed Switch Indication System~~.

Insert 3

SR 3.1.4.6

Verification of CEA drop times determined that the maximum CEA drop time permitted is consistent with the assumed drop time used in that safety analysis (Reference 1, Chapter 14). Control element assembly drop time is measured from the time when electrical power is interrupted to the CEDM until the

BASES

reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown CEAs are withdrawn before the regulating CEAs are withdrawn during a unit startup.

Insert 3

~~Since the shutdown CEAs are positioned manually by the Control Room operator, verification of shutdown CEA position at a Frequency of 12 hours is adequate to ensure that the shutdown CEAs are within their insertion limits. Also, the 12 hour Frequency takes into account other information available to the operator in the Control Room for the purpose of monitoring the status of the shutdown CEAs.~~

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

BASES

ensures improper CEA alignments are identified before unacceptable flux distributions occur.

E.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

With the PDIL alarm circuit OPERABLE, verification of each regulating CEA group position ~~every 12 hours~~ is sufficient to detect CEA positions that may approach the acceptable limits, and to provide the operator with time to undertake the Required Action(s) should the sequence or insertion limits be found to be exceeded. ~~The 12-hour Frequency also takes into account the indication provided by the PDIL alarm circuit and other information about CEA group positions available to the operator in the Control Room.~~

Insert 3

SR 3.1.6.2

Verification of the accumulated time of CEA group insertion between the long-term steady state insertion limits and the transient insertion limits ensures the cumulative time limits are not exceeded. ~~The 24-hour Frequency ensures the operator identifies a time limit that is being approached before it is reached.~~

SR 3.1.6.3

Demonstrating the PDIL alarm circuit OPERABLE verifies that the PDIL alarm circuit is functional. ~~The 31-day Frequency takes into account other SRs being performed at shorter Frequencies that identify improper CEA alignments.~~

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" 10 CFR 50.46

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification of the position of each partially or fully withdrawn full-length or part-length CEA is necessary to ensure that the minimum negative reactivity requirements for insertion on a trip are preserved. ~~A 2-hour Frequency is sufficient for the operator to verify that each CEA position is within the acceptance criteria.~~

Insert 3

SR 3.1.7.2

Prior demonstration that each CEA to be withdrawn from the core during PHYSICS TESTS is capable of full insertion, when tripped from at least a 50% withdrawn position, ensures that the CEA will insert on a trip signal. The Frequency ensures that the CEAs are OPERABLE prior to reducing SDM to less than the limits of LCO 3.1.1.

The SR is modified by a Note that allows the SR to not be performed during initial power escalation following a refueling outage if SR 3.1.4.6 has been met during that refueling outage. This allows the CEA drop time test, which also proves the CEAs are trippable, to be credited for this SR.

REFERENCES

1. 10 CFR Part 50
2. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978
3. UFSAR

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verifying that THERMAL POWER is equal to or less than that allowed by the test power plateau, as specified in the PHYSICS TESTS procedure and required by the safety analysis, ensures that adequate LHR and DNB parameter margins are maintained while LCOs are suspended. ~~The 1 hour Frequency is sufficient, based on the slow rate of power change and increased operational controls in place during PHYSICS TESTS.~~

Inert 3

REFERENCES

1. 10 CFR Part 50
2. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978
3. UFSAR

BASES

is not within limits. Therefore, this SR is only applicable when the Excore Detector Monitoring System is being used to determine the LHR. ~~The 31-day Frequency is appropriate for this SR because it is consistent with the requirements of SR 3.3.1.3 for calibration of the excore detectors using the incore detectors.~~

Insert 3

The SR is modified by a Note that states that the SR is only applicable when the Excore Detection Monitoring System is being used to determine LHR. The reason for the Note is that the excore detectors input neutron flux information into the ASI calculation.

SR 3.2.1.3 and SR 3.2.1.4

Continuous monitoring of the LHR is provided by the Incore Detector Monitoring System and the Excore Detector Monitoring System. Either of these two core power distribution monitoring systems provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its specified limits.

Performance of these SRs verifies that the Incore Detector Monitoring System can accurately monitor LHR. Therefore, they are only applicable when the Incore Detector Monitoring System is being used to determine the LHR.

Insert 3

~~A 31-day Frequency is consistent with the historical testing frequency of the incore detector monitoring system.~~ The SRs are modified by two Notes. Note 1 allows the SRs to be performed only when the Incore Detector Monitoring System is being used to determine LHR. Note 2 states that the SRs are not required to be performed when THERMAL POWER is < 20% RTP. The accuracy of the neutron flux information from the incore detectors is not reliable at THERMAL POWER < 20% RTP.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR)
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The periodic SR to determine the calculated F_r^T ensures that F_r^T remains within the range assumed in the analysis throughout the fuel cycle. Determining the measured F_r^T once after each fuel loading prior to exceeding 70% RTP ensures that the core is properly loaded.

Insert 3

~~Performance of the SR every 31 days of accumulated operation in MODE 1 provides reasonable assurance that unacceptable changes in the F_r^T are promptly detected.~~

The power distribution map can only be obtained after THERMAL POWER exceeds 20% RTP because the incore detectors are not reliable below 20% RTP.

The SR is modified by a Note that requires the incore detectors to be used to determine F_r^T by using them to obtain a power distribution map with all full length CEAs above the long-term steady state insertion limits, as specified in the COLR.

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

BASES

is necessary to account explicitly for power asymmetries because the radial power peaking factor used in core power distribution calculations is based on an untilted power distribution.

If T_q is not restored to within its limits, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation that causes increased LHRs when the xenon redistributes. If T_q cannot be restored to within its limits within 2 hours, reactor power must be reduced.

C.1

If Required Actions and associated Completion Times of Condition A or B are not met, THERMAL POWER must be reduced to ≤ 50% RTP. This requirement provides conservative protection from increased peaking due to potential xenon redistribution and provides reasonable assurance that the core is operating within its thermal limits and places the core in a conservative condition. Four hours is a reasonable time to reach 50% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

~~T_q must be calculated at 12 hour intervals.~~ T_q is determined using the incore and excore detectors. When one excore channel is inoperable and THERMAL POWER is > 75% RTP, the incore detectors shall be used. ~~The 12-hour Frequency prevents significant xenon redistribution between surveillance tests.~~

Insert 3

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

BASES

ACTIONS

A.1

Operating the core within ASI limits specified in the COLR and within the limits of LCO 3.3.1 ensures an acceptable margin for DNB and for maintaining local power density in the event of an AOO. Maintaining ASI within limits also ensures that the limits of Reference 2 are not exceeded during accidents. The Required Actions to restore ASI must be completed within 2 hours to limit the duration the plant is operated outside the initial conditions assumed in the accident analyses. In addition, this Completion Time is sufficiently short that the xenon distribution in the core cannot change significantly.

B.1

If the ASI cannot be restored to within its specified limits, or ASI cannot be determined because of Excore Detector Monitoring System inoperability, core power must be reduced. Reducing THERMAL POWER to ≤ 20% RTP provides reasonable assurance that the core is operating farther from thermal limits and places the core in a conservative condition. Four hours is a reasonable amount of time, based on operating experience, to reduce THERMAL POWER to ≤ 20% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

Verifying that the ASI is within the specified limits provides reasonable assurance that the core is not approaching DNB conditions. ~~A Frequency of 12 hours is adequate for the operator to identify trends in conditions that result in an approach to the ASI limits, because the mechanisms that affect the ASI, such as xenon redistribution or CEA drive mechanism malfunctions, cause the ASI to change slowly and should be discovered before the limits are exceeded.~~

Insert 3

BASES

SURVEILLANCE REQUIREMENTS

The SRs for any particular RPS Function are found in the SR column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.1.1

Performance of the CHANNEL CHECK ~~once every 12 hours~~ ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one instrument channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument channel drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a qualitative assessment of the instrument channel combined with the instrument channel uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits. CHANNEL CHECKS are performed on the wide range logarithmic neutron flux monitor for the Rate of Change of Power-High trip Function.

Insert 3

~~The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of instrument channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of RPS Function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of the channel during normal operational use of the displays.~~

BASES

SR 3.3.1.2

A daily calibration (heat balance) is performed when THERMAL POWER is ≥ 15%. The daily calibration shall consist of adjusting the "nuclear power calibrate" potentiometers to agree with the calorimetric calculation if the absolute difference is > 1.5%. The "ΔT power calibrate" potentiometers are then used to null the "nuclear power-ΔT power" indicators on the RPS Calibration and Indication Panel. Performance of the daily calibration ensures that the two inputs to the Q power measurement are indicating accurately with respect to the much more accurate secondary calorimetric calculation. The heat balance addresses overall gain of the instruments and does not include ASI.

Insert 3

~~The Frequency of 24 hours is based on plant operating experience and takes into account indications and alarms located in the Control Room to detect deviations in channel outputs.~~ The Frequency is modified by a Note indicating that once the unit reaches 15% RTP, 12 hours is the maximum time allowed for completing this Surveillance. The secondary calorimetric is inaccurate at lower power levels. The 12 hours allows time for plant stabilization, data-taking, and instrument calibration.

A second Note indicates the daily calibration may be suspended during PHYSICS TESTS. This ensures that calibration is proper both preceding and following physics testing at each plateau, recognizing that during testing, changes in power distribution and RCS temperature may render the calibration inaccurate.

SR 3.3.1.3

It is necessary to calibrate the excore power range channel upper and lower subchannel amplifiers such that the internal ASI used in the TM/LP trip and APD-High trip Functions reflects the true core power distribution as determined by the incore detectors. A Note indicates that once the unit reaches 20% RTP, 12 hours is the maximum time allowed for completion of this Surveillance. The Surveillance is required to be performed prior to operation above 90% RTP. Uncertainties in the excore and incore measurement process make it impractical to calibrate when THERMAL POWER is

BASES

< 20% RTP. The Completion Time of 12 hours allows time for plant stabilization, data-taking, and instrument calibration. ~~The Frequency requires the SR be performed every 31 days after the initial performance prior to operation above 90% RTP.~~ Requiring the SR prior to operations above 90% RTP is because of the increased uncertainties associated with using uncalibrated excore detectors. If the excore channels are not properly calibrated to agree with the incore detectors, power is restricted during subsequent operations because of increased uncertainty associated with using uncalibrated excore channels. ~~The 31 day Frequency is adequate, based on operating experience of the excore linear amplifiers and the slow burnup of the detectors. The excore readings are a strong function of the power produced in the peripheral fuel bundles and do not represent an integrated reading across the core. Slow changes in neutron flux during the fuel cycle can also be detected at this Frequency.~~

Insert 3

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed on each RPS instrument channel, except Loss of Load and Rate of Change of Power, ~~every 92 days~~ to ensure the entire channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

In addition to reference voltage power supply tests, the RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference 1, Section 7.2. These tests verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. They include:

Bistable Tests

The bistable setpoint must be found to trip within the Allowable Values specified in the LCO and left set consistent with the assumptions of Reference 4. As-found values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference 8.

A test signal is substituted as the input in one instrument channel at a time to verify that the bistable trip unit trips within the specified tolerance around the setpoint. This is done with the affected RPS channel bistable trip unit bypassed. Any setpoint adjustment shall be consistent with the assumptions of Reference 4.

Matrix Logic Tests

Matrix logic tests are addressed in LCO 3.3.3. This test is performed one matrix at a time. It verifies that a coincidence in the two instrument channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. This test will detect any short circuits around the bistable contacts in the coincidence logic, such as may be caused by faulty bistable relay or trip bypass contacts.

Trip Path Tests

Trip path logic tests are addressed in LCO 3.3.3. These tests are similar to the matrix logic tests, except that test power is withheld from one matrix relay at a time, allowing the trip path circuit to de-energize, opening the affected set of RTCBs. The RTCBs must then be closed prior to testing the other three trip path circuits, or a reactor trip may result.

Insert 3

~~The Frequency of 92 days is based on the reliability analysis presented in Reference 6.~~

SR 3.3.1.5

A CHANNEL CALIBRATION of the excore power range channels ~~every 92 days~~ ensures that the channels are reading accurately and within tolerance. The SR verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves

BASES

the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant-specific SRs.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the Frequency extension analysis. The requirements for this review are outlined in Reference 8.

A Note is added stating that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal (Reference 7). Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.2) and the monthly linear subchannel gain check (SR 3.3.1.3). In addition, associated control room indications are continuously monitored by the operators.

Insert 3

~~The Frequency of 92 days is acceptable, based on plant operating experience, and takes into account indications and alarms available to the operator in the Control Room.~~

SR 3.3.1.6

A CHANNEL FUNCTIONAL TEST on the Loss of Load, and Rate of Change of Power channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function if required. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. The Loss of Load sensor cannot be tested during reactor operation without causing reactor trip. The Power Rate of Change-High trip Function is required during startup operation and is bypassed when shut down or > 12% RTP.

BASES

SR 3.3.1.7

Surveillance Requirement 3.3.1.7 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.1.4, except SR 3.3.1.7 is applicable only to Functions with automatic bypass removal features. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. Proper operation of operating bypasses are critical during plant startup because the bypasses must be in place to allow startup operation and must be removed at the appropriate points during power ascent to enable certain reactor trips. Insert 3 ~~A 24 month SR Frequency is adequate to ensure proper automatic bypass removal feature operation as described in Reference 5.~~ Once the operating bypasses are removed, the bypasses must not fail in such a way that the associated trip Function gets inadvertently bypassed. This feature is verified by the trip Function CHANNEL FUNCTIONAL TEST, SR 3.3.1.4. Therefore, further testing of the automatic bypass removal feature after startup is unnecessary.

SR 3.3.1.8

Surveillance Requirement 3.3.1.8 is the performance of a CHANNEL CALIBRATION ~~every 24 months~~.

CHANNEL CALIBRATION is a check of the instrument channel, including the sensor. The SR verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument channel drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 4.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the frequency extension analysis. The requirements for this review are outlined in Reference 6.

Insert 3

~~The Frequency is based upon the assumption of a 24-month calibration interval for the determination of the magnitude of equipment drift.~~

The SR is modified by a Note to indicate that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift, and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the ~~daily~~ calorimetric calibration (SR 3.3.1.2) and the ~~monthly~~ linear subchannel gain check (SR 3.3.1.3).

SR 3.3.1.9

This SR ensures that the RPS RESPONSE TIMES are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the RTCBs open. ~~Response times are conducted on a 24-month STAGGERED TEST BASIS.~~ Response time testing acceptance criteria are included in Reference 1, Section 7.2. ~~This results in the interval between successive SRs of a given channel of n x 24 months, where n is the number of channels in the function. The Frequency of 24 months is based upon operating experience, which has shown that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. Also, response times cannot be determined at power since equipment operation is required.~~ Testing may be performed in one measurement or in overlapping segments, with verification that all components are tested.

Insert 3

Response time may be verified by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, such that the response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. Reference 9 provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response

D.1, D.2.1, and D.2.2

Condition D applies to two inoperable automatic bypass removal features. If the automatic bypass removal features cannot be restored to OPERABLE status, the associated Rate of Change of Power-High trip RPS channel may be considered OPERABLE only if the bypasses are not in effect. Otherwise, the affected RPS channels must be declared inoperable, as in Condition B, and the bypasses either removed or the automatic bypass removal features repaired. Also, Required Action D.2.2 provides for the restoration of the one affected automatic trip channel to OPERABLE status within the rules of Completion Time specified under Condition B. Completion Times are consistent with Condition B.

E.1

Condition E is entered when the Required Actions and associated Completion Times of Condition A, B, C, or D are not met.

If Required Actions associated with these Conditions cannot be completed within the required Completion Time, opening the RTCBs brings the reactor to a MODE where the LCO does not apply and ensures no CEA withdrawal will occur. The basis for the Completion Time of 6 hours is that it is adequate to complete the Required Actions without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

Performance of the CHANNEL CHECK on each wide range channel ~~once every 12 hours~~ ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one instrument channel to a similar parameter on another channel. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument channel drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying

BASES

that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a qualitative assessment of the instrument channel that considers instrument channel uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

Insert 3

~~The Frequency, once every shift, is based on operating experience that demonstrates the rarity of instrument channel failure. Since the probability of two random failures in redundant channels in any 12-hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of RPS Function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of the channel during normal operational use of the displays.~~

SR 3.3.2.2

A CHANNEL FUNCTIONAL TEST on the power rate of change channels is performed once within 7 days prior to each reactor startup to ensure the entire instrument channel will perform its intended function if required. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. The Rate of Change of Power-High trip Function is required during startup operation and is bypassed when shut down or > 12% RTP. Additionally, operating experience has shown that these components usually pass the SR when performed at a Frequency of once within 7 days prior to each reactor startup.

Only the Allowable Values are specified for each RPS trip Function in the SR. Nominal trip setpoints are established for the Functions via the plant-specific procedures. The nominal setpoints are selected to ensure the plant parameters do not exceed the Allowable Value if the bistable trip unit is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant-specific setpoint calculations. Each nominal trip setpoint is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument channel uncertainties appropriate to the trip Function. These uncertainties are defined in Reference 3.

SR 3.3.2.3

Surveillance Requirement 3.3.2.3 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.2.2, except SR 3.3.2.3 is applicable only to bypass Functions ~~and is performed once every 24 months~~. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

Proper operation of operating bypasses is critical during plant startup because the bypasses must be in place to allow startup operation and must be removed at the appropriate points during power ascent to enable certain reactor trips.

Insert 3

~~A 24-month SR Frequency is adequate to ensure proper automatic bypass removal feature operation as described in Reference 5.~~ Once the operating bypasses are removed, the bypasses must not fail in such a way that the associated trip Function gets inadvertently bypassed. This feature is verified by SR 3.3.2.2. Therefore, further testing of the automatic bypass removal feature after startup is unnecessary.

SR 3.3.2.4

Surveillance Requirement 3.3.2.4 is the performance of a CHANNEL CALIBRATION ~~every 24 months~~.

BASES

CHANNEL CALIBRATION is a check of the instrument channel including the sensor. The SR verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 3.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the SR interval extension analysis. The requirements for this review are outlined in Reference 4.

Insert 3

~~The Frequency is based upon the assumption of a 24-month calibration interval in the determination of the magnitude of equipment drift.~~

The SR is modified by a Note to indicate that the neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices with minimal drift (Reference 5).

REFERENCES

1. 10 CFR Parts 50, "Domestic Licensing of Production and Utilization Facilities," and 100, "Reactor Site Criteria"
2. Updated Final Safety Analysis Report (UFSAR), Chapter 14, "Safety Analysis"
3. CCNPP Setpoint File
4. Combustion Engineering Topical Report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" dated June 2, 1986, including Supplement 1, March 3, 1989
5. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 6, 1995, "License Amendment Request; Extension of Instrument Surveillance Intervals"

BASES

one manual trip, matrix logic, trip path logic, or RTCB channel is inoperable for reasons other than Condition A or D.

If the RTCBs associated with the inoperable channel cannot be opened, the reactor must be shut down within 6 hours and all the RTCBs opened. A Completion Time of 6 hours is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner, without challenging plant systems, and to open RTCBs. All RTCBs should then be opened, placing the plant in a MODE where the LCO does not apply and ensuring no CEA withdrawal occurs.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

A CHANNEL FUNCTIONAL TEST is performed on each RTCB channel ~~every 92 days~~. This verifies proper operation of each RTCB. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. The RTCB must then be closed prior to testing the other RTCBs, or a reactor trip may result. ~~The frequency of 92 days is based on the reliability analysis presented in Reference 3. Scheduling SR 3.3.3.1 and SR 3.3.3.2 such that the RTCBs testing is performed at least every 6 weeks meets vendor recommended intervals for cycling of each RTCB in accordance with Reference 3.~~

Insert 3

SR 3.3.3.2

A CHANNEL FUNCTIONAL TEST on each RPS logic channel is performed ~~every 92 days~~ to ensure the entire channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at

BASES

least once per refueling interval with applicable extensions.

In addition to reference voltage tests, the RPS CHANNEL FUNCTIONAL TEST consists of three overlapping tests as described in Reference 1, Section 7.2. These tests verify that the RPS is capable of performing its intended function, from bistable input through the RTCBs. The first test, the instrument channel test, is addressed by SR 3.3.1.4 in LCO 3.3.1.

This SR addresses the two tests associated with the RPS logic: matrix logic and trip path logic.

~~Scheduling SR 3.3.3.1 and SR 3.3.3.2 such that the RTCBs testing is performed at least every 6 weeks meets vendor recommended intervals for cycling of each RTCB in accordance with Reference 3.~~

Matrix Logic Tests

These tests are performed one matrix at a time. They verify that a coincidence in the two instrument channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. The matrix logic tests will detect any short circuits around the bistable contacts in the coincidence logic such as may be caused by faulty bistable relay or trip bypass contacts.

Trip Path Tests

These tests are similar to the matrix logic tests, except that test power is withheld from one matrix relay at a time, allowing the trip path circuit to de-energize, opening the affected set of RTCBs. The RTCBs must then be closed prior to testing the other three trip path circuits, or a reactor trip may result.

Insert 3

~~The Frequency of 92 days is based on the reliability analysis presented in Reference 4.~~

BASES

SURVEILLANCE REQUIREMENTS

The SRs for any particular ESFAS Function are found in the SRs column of Table 3.3.4-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, CHANNEL CALIBRATION, and response time testing.

SR 3.3.4.1

Performance of the CHANNEL CHECK ~~once every 12 hours~~ ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one sensor channel to a similar parameter on other sensor channels. It is based on the assumption that sensor channels monitoring the same parameter should read approximately the same value. Significant deviations between sensor channels could be an indication of excessive sensor channel drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a qualitative assessment of the sensor channel, which considers sensor channel uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off-scale during times when surveillance testing is required, the CHANNEL CHECK will only verify that they are off-scale in the same direction. Off-scale low current loop channels are verified to be reading at the bottom of the range and not failed down-scale.

Insert 3

~~The Frequency of about once every shift is based on operating experience that demonstrates sensor channel failure is rare. Since the probability of two random failures in redundant channels in any 12-hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of ESFAS Function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of the channel during normal operational use of displays.~~

SR 3.3.4.2

Insert 3

A CHANNEL FUNCTIONAL TEST is performed ~~every 92 days~~ to ensure the entire sensor channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

The CHANNEL FUNCTIONAL TEST tests the individual sensor channels using an analog or level switch test input to each bistable.

A test signal is substituted for the input in one sensor channel at a time to verify that the bistable trips within the specified tolerance around the setpoint. Any setpoint adjustment shall be consistent with the assumptions of the Reference 5.

SR 3.3.4.3

Surveillance Requirement 3.3.4.3 is a CHANNEL FUNCTIONAL TEST similar to SR 3.3.4.2, except 3.3.4.3 is ~~performed every 24 months and is~~ only applicable to automatic block removal features of the sensor block modules. These include the Pressurizer Pressure-Low trip block and the SGIS Steam Generator Pressure-Low trip block. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

The CHANNEL FUNCTIONAL TEST for proper operation of the automatic block removal features is critical during plant heatups because the blocks may be in place prior to entering MODE 3, but must be removed at the appropriate points during plant startup to enable the ESFAS Function. ~~A 24-month SR~~

BASES

~~Frequency is adequate to ensure proper automatic block removal module operation, as described in Reference 3.~~ Once the blocks are removed, the blocks must not fail in such a way that the associated ESFAS Function is inappropriately blocked. This feature is verified by the appropriate ESFAS Function CHANNEL FUNCTIONAL TEST.

Insert 3

~~The 24-month SR Frequency is adequate to ensure proper automatic block removal feature operation as described in Reference 3.~~

SR 3.3.4.4

CHANNEL CALIBRATION is a check of the sensor channel, including the automatic block removal feature of the sensor block module and the sensor. The SR verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for sensor channel drift between successive calibrations to ensure that the channel remains operational between successive surveillance tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 5.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the extension analysis. The requirements for this review are outlined in Reference 6.

Insert 3

~~The Frequency is based upon the assumption of a 24-month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.~~

SR 3.3.4.5

This SR ensures that the train actuation response times are the maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. The analysis models the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment in both trains reaches the required functional state (e.g., pumps are rated discharge pressure, valves in full open or closed position). Response time testing acceptance criteria are included in Reference 1, Section 7.3. The test may be performed in one measurement or in overlapping segments, which verification that all components are measured.

Response time may be verified by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, such that the response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. Reference 7 provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the reference. Response time verification for other sensor types must be demonstrated by test. The allocation of sensor response times must be verified prior to placing a new component in operation and reverified after maintenance that may adversely affect the sensor response time.

Instrument loop or test cables and wiring add an insignificant response time and can be ignored.

Insert 3

~~Engineered Safety Feature Response Time tests are conducted on a STAGGERED TEST BASIS of once every 24 months. This results in the interval between successive tests of a given channel of n x 24 months, where n is the number of channels in the Function. Surveillance of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in Response Time verification of these devices every 24 months. The 24-month STAGGERED TEST BASIS Frequency is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~

REFERENCES

1. UFSAR
2. IEEE No. 279, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," August 1968

BASES

which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

A CHANNEL FUNCTIONAL TEST is performed ~~every 92 days~~ to ensure the entire actuation logic channel will perform its intended function when needed. Sensor channel tests are addressed in LCO 3.3.4. This SR addresses actuation logic tests. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

Actuation Logic Tests

Actuation logic channel testing includes injecting one actuation signal into each two-out-of-four logic actuation modules in each ESFAS Function, and using a bistable trip input to satisfy the actuation logic. Testing includes block logic modules.

Note 1 requires that actuation logic tests include operation of actuation relays. Note 2 allows deferred at power testing of certain subchannel relays to allow for the fact that operating certain relays during power operation could cause plant transients or equipment damage. Those subchannel relays that cannot be tested at power must be tested in accordance with Note 2. These include SIAS No. 5, SIAS No. 10, CIS No. 5, SGIS No. 1, and CSAS No. 3.

These subchannel relays actuate the following components, which cannot be tested at power:

- RCP seal bleedoff isolation valves;
- Service water isolation valves;

- Volume control tank discharge valves;
- Letdown stop valves;
- Component Cooling to and from the RCPs;
- MSIVs and feedwater isolation valves;
- Instrument air CIVs;
- Heater drain pumps;
- Main feedwater pumps; and
- Condensate booster pumps.

The reasons each of the above cannot be fully tested at power are stated in Reference 1.

Actuation logic tests verify that the ESFAS is capable of performing its intended function, from bistable input through the actuated components.

Insert 3

~~The Frequency of 92 days is based on operating experience that has shown these components usually pass the surveillance test when performed at this Frequency.~~

SR 3.3.5.2

A CHANNEL FUNCTIONAL TEST is performed on the manual ESFAS actuation circuitry, de-energizing relays and providing manual actuation of the Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

This surveillance test verifies that the actuation push buttons are capable of opening contacts in the actuation logic as designed, de-energizing the actuation relays and providing manual trip of the Function. ~~The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a plant outage, and the potential for an unplanned transient if the test were to be performed with the reactor at power. Operating~~

Insert 3

BASES

~~experience has shown these components usually pass the surveillance test when performed at a Frequency of once every 24 months.~~

REFERENCES

1. UFSAR, Section 7.3, "Engineered Safety Features Actuation Systems"
2. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 5, 1995, "Response to NRC Request for Review & Comment on Review of Preliminary Accident Precursor Analysis of Trip; Loss of 13.8 kV Bus; Short-Term Saltwater Cooling System Unavailability, CCNPP Unit 2"

degraded condition in an orderly manner and takes into account the low probability of an event requiring LOVS occurring during this interval.

D.1

Condition D applies if the Required Actions and associated Completion Times are not met.

Required Action D.1 ensures that Required Actions for the affected DG inoperabilities are initiated. The actions specified in LCO 3.8.1 are required immediately.

SURVEILLANCE REQUIREMENTS

The following SRs apply to each DG-LOVS Function.

SR 3.3.6.1

A CHANNEL FUNCTIONAL TEST is performed ~~every 92 days~~ to ensure that the entire sensor channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

Insert 3

~~The Frequency of 92 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one sensor channel of a given function in any 92 day Frequency is a rare event. Any setting adjustment shall be consistent with the assumptions of the current plant specific setting analysis.~~

SR 3.3.6.2

Surveillance Requirement 3.3.6.2 is the performance of a CHANNEL CALIBRATION ~~every 24 months~~. The CHANNEL CALIBRATION verifies the accuracy of each component within the sensor channel, except stepdown transformers, which are not calibrated. This includes calibration of the undervoltage relays and demonstrates that the equipment falls within the specified operating characteristics defined by the manufacturer.

The SR verifies that the sensor channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the plant-specific setting analysis.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the SR interval extension analysis. The requirements for this review are outlined in Reference 4.

The settings, as well as the response to Loss of Voltage and Degraded Voltage tests, shall include a single point verification that the trip occurs within the required delay time as shown in Reference 1, Section 7.3. ~~The Frequency is based upon the assumption of a 24 month calibration interval for the determination of the magnitude of equipment drift in the plant setting analyses~~.

Insert 3

REFERENCES

1. UFSAR
2. CCNPP Setpoint File
3. IEEE No. 279, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," August 1968
4. Calvert Cliffs Procedure EN-4-104, "Surveillance Testing"

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

Performance of the CHANNEL CHECK ~~once every 12 hours~~ ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one sensor channel to a similar parameter on other channels. It is based on the assumption that sensor channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two sensor channels could be an indication of excessive sensor channel drift in one of the channels or of something more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a qualitative assessment of the sensor channel that considers sensor channel uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits.

Insert 3

~~The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of sensor channel failure. Since the probability of two random failures in redundant channels in any 12-hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of the channel during normal operational use of the displays.~~

SR 3.3.7.2

Proper operation of the actuation relays is verified by verification of the relay driver output signal. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

~~The Frequency of 92 days is based on plant operating experience with regard to actuation channel OPERABILITY, which demonstrates that failure of more than one channel of a given Function in any 92-day interval is a rare event.~~

Insert 3

SR 3.3.7.3

A CHANNEL FUNCTIONAL TEST is performed on each containment radiation sensor channel to ensure the entire channel, except for sensor and initiating relays, will perform its intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

~~The Frequency of 92 days is based on plant operating experience with regard to sensor channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 92 day interval is a rare event.~~

SR 3.3.7.4

CHANNEL CALIBRATION is a check of the sensor channel including the sensor. The Surveillance verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for sensor channel drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 2.

~~The Frequency is based upon the assumption of a 24 month calibration interval based on the refueling interval and the instruments not being inservice during power operations, but part of preparation for being placed in service is a CHANNEL CALIBRATION.~~

SR 3.3.7.5

~~Every 24 months,~~ a CHANNEL FUNCTIONAL TEST is performed on the manual CRS actuation circuitry. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

This surveillance test verifies that the actuation push buttons are capable of opening contacts in the actuation logic as designed, de-energizing the actuation relays and providing manual actuation of the Function. ~~The 24-month Frequency is based on the need to perform this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if the SR were performed with the reactor at power. Operating experience has shown these components usually pass the surveillance test when performed at a Frequency of once every 24 months.~~

Insert 3

SR 3.3.7.6

This surveillance test ensures that the train actuation response times are less than or equal to the maximum times assumed in the analyses. Response times are defined in the same manner as ESF RESPONSE TIME. Response time testing acceptance criteria are included in Reference 1, Section 7.3. ~~The 24-month Frequency is based upon plant operating experience, which shows random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~ Testing of the final actuating devices, which make up the bulk of the response time, is included. Testing of the final actuating device is one channel is included in the testing of each actuation logic channel.

Insert 3

REFERENCES

1. UFSAR
2. CCNPP Setpoint File

A.1, B.1, B.2, C.1, C.2.1, and C.2.2

Conditions A, B, and C are applicable to the CRRS trip circuit and measurement channel. Condition A applies to the failure of the CRRS trip circuit or measurement channel in MODE 1, 2, 3, or 4. Entry into this Condition requires action to either restore the failed channel or manually perform the CREVS function (Required Action A.1). The Completion Time of 1 hour is sufficient to complete the Required Actions. If the channel cannot be restored to OPERABLE status, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The Completion Times of 6 hours and 36 hours for reaching MODEs 3 and 5 from MODE 1 are reasonable, based on operating experience and normal cooldown rates, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant safety systems or operators.

Condition C applies to the failure of the CRRS trip circuit or measurement channel when moving irradiated assemblies. The Required Actions are immediately taken to place one OPERABLE CREVS train in the recirculation mode with post-LOCA fans in service or to suspend movement of irradiated fuel assemblies. The Completion Time recognizes the fact that the radiation signal is the only Function available to initiate control room isolation in the event of a Fuel Handling Accident.

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

Performance of the CHANNEL CHECK ~~once every 12 hours~~ ensures that a gross failure of instrumentation has not occurred. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Acceptance criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit.

Insert 3

~~The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure~~. ~~The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels. In addition, a down-scale alarm and up-scale alarm immediately alert operations to loss of the channel~~.

SR 3.3.8.2

A CHANNEL FUNCTIONAL TEST is performed on the control room radiation monitoring channel to ensure the entire channel will perform its intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

Insert 3

~~The Frequency of 92 days is based on plant operating experience with regard to channel OPERABILITY and drift.~~

SR 3.3.8.3

CHANNEL CALIBRATION is a check of the CRRS channel, including the sensor. The surveillance test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for channel drift between successive calibrations to ensure that the channel remains operational between successive surveillance tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 2.

~~The Frequency of 24 months has been shown by operating experience to be adequate to detect any failures.~~

REFERENCES

1. UFSAR
2. CCNPP Setpoint File

Restoring at least one sensor channel to OPERABLE status is the preferred Required Action. If this cannot be accomplished, one channel should be placed in bypass and the other channel in trip. The allowed Completion Time of 1 hour is sufficient time to perform the Required Actions.

Once the Required Action to trip or bypass the channel has been complied with, Required Action C.2 provides for restoring one channel to OPERABLE status within 48 hours. The justification of the 48-hour Completion Time is the same as for Condition B.

After one channel is restored to OPERABLE status, the provisions of Condition C still apply to the remaining inoperable channel.

D.1 and D.2

Condition D specifies the shutdown track to be followed if the Required Actions and associated Completion Times of Condition A, B, or C are not met. If the Required Actions cannot be met within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK on each CVCS isolation pressure indicating sensor channel ~~once every 12 hours~~ ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that sensor channels monitoring the same parameter should read approximately the same value.

Significant deviations between the two sensor channels could be an indication of excessive sensor channel drift in one of the channels or of something more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a qualitative assessment of the sensor channel that considers sensor channel uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit.

~~The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12-hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays.~~

Insert 3

SR 3.3.9.2

A CHANNEL FUNCTIONAL TEST is performed on each sensor channel to ensure the entire channel, except for sensor and initiation logic, will perform its intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions.

~~The Frequency of 92 days is based on plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 92-day interval is a rare event.~~

Note 1 indicates proper operation of the individual actuation relays is verified by verification of proper relay driver output signal. Note 2 indicates that relays that cannot be tested at power are excepted from the SR while at power. These relays must, however, be tested once per 24 months.

SR 3.3.9.3

CHANNEL CALIBRATION is a check of the sensor channel including the sensor. The surveillance test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for sensor channel drift between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with Reference 2.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the SR interval extension analysis. The requirements for this review are outlined in Reference 4.

Radiation detectors may be removed and calibrated in a laboratory, calibrated in place using a transfer source, or replaced with an equivalent laboratory calibrated unit.

~~The Frequency is based upon the assumptions of a 24-month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis, and includes operating experience, as well as consistency with a 24-month fuel cycle~~.

Insert 3

SR 3.3.9.4

This surveillance test ensures that the train actuation response times are less than or equal to the maximum times assumed in the analyses. Response times are defined in the same manner as ESF RESPONSE TIME. ~~The 24-month Frequency is based upon plant operating experience, which shows random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences~~. Testing of the final actuating devices, which make up the bulk of the response time, is included. Testing of the final actuating device in one channel is included in the testing of each actuation logic channel.

SR 3.3.10.1

Performance of the CHANNEL CHECK ~~once every 31 days~~ ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one indication channel to a similar parameter on other channels. It is based on the assumption that indication channels monitoring the same parameter should read approximately the same value. Significant deviations between the two indication channels could be an indication of excessive instrument drift in one of the channels or of something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a qualitative assessment of the indication channel that considers indication channel uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off-scale during times when surveillance testing is required, the CHANNEL CHECK will only verify that they are off-scale in the same direction. Off-scale low current loop channels are verified to be reading at the bottom of the range and not failed down-scale.

Insert 3

~~The Frequency of 31 days is based upon plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one indication channel of a given Function in any 31 day interval is a rare event. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel during normal operational use of the displays associated with this LCO's required channels.~~

SR 3.3.10.2

Deleted.

SR 3.3.10.3

A ~~CHANNEL CALIBRATION is performed every 24 months or approximately every refueling.~~ CHANNEL CALIBRATION is a check of the indication channel including the sensor. The SR verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION of the CIV position indication channels will consist of verification that the position indication changes from not-closed to closed when the valve is exercised to the isolation position as required by Technical Specification 5.5.8, Inservice Testing Program. The position switch is the sensor for the CIV position indication channels. A Note allows exclusion of neutron detectors, CETs, and reactor vessel level (HJTC) from the CHANNEL CALIBRATION.

Insert 3

~~The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is justified by an 24 month calibration interval for the determination of the magnitude of equipment drift.~~

REFERENCES

1. Letter from Mr. R. E. Denton (BGE) to NRC Document Control Desk, dated June 6, 1995, "License Amendment Request; Extension of Instrument Surveillance Intervals"
2. Letter from Mr. J. A. Tiernan (BGE) to NRC Document Control Desk, dated August 9, 1988, "Regulatory Guide 1.97 Review Update"
3. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident (Errata Published July 1981), December 1975
4. NUREG-0737, Supplement 1, Requirements for Emergency Response Capabilities (Generic Letter 82-33), December 17, 1982
5. UFSAR, Chapter 7, "Instrumentation and Control"

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1

Performance of the CHANNEL CHECK ~~once every 31 days~~ ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one indication channel to a similar parameter on other channels. It is based on the assumption that indication channels monitoring the same parameter should read approximately the same value. Significant deviations between the indication channels could be an indication of excessive instrument drift in one of the channels or of something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. Agreement criteria are determined by the plant staff, based on a qualitative assessment of the indication channel that considers indication channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off-scale during times when surveillance testing is required, the CHANNEL CHECK will only verify that they are off-scale in the same direction. Off-scale low current loop channels are verified to be reading at the bottom of the range and not failed down-scale.

Insert 3

~~The Frequency is based on plant operating experience that demonstrates indication channel failure is rare~~.

SR 3.3.11.2

CHANNEL CALIBRATION is a check of the indication channel including the sensor. The surveillance test verifies that the channel responds to the measured parameter within the necessary range and accuracy.

~~The 24-month Frequency is based upon the need to perform this SR under the conditions that apply during a plant outage, and the potential for an unplanned transient if the~~

BASES

~~surveillance test were to be performed with the reactor at power~~.

The SR is modified by a Note, which excludes neutron detectors and reactor trip breaker indication from the CHANNEL CALIBRATION.

REFERENCES

1. Updated Final Safety Analysis Report, Appendix 1C, "AEC Proposed General Design Criteria for Nuclear Power Plants"

reduced. These Completion Times are also based on operating experience in performing the Required Actions and the fact that plant conditions will change slowly.

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

Surveillance Requirement 3.3.12.1 is the performance of a CHANNEL CHECK on each required channel ~~every 12 hours~~. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based upon the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of something more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff and should be based on a qualitative assessment of the indication channel that considers indication channel uncertainties, including control isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limits. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

Insert 3

~~The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12-hour period is extremely low, CHANNEL CHECK minimizes the chance of loss of indication due to failure of redundant channels. CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of displays associated with the LCO required channels.~~

BASES

SR 3.3.12.2

A CHANNEL FUNCTIONAL TEST is performed once within 7 days prior to each reactor startup. This SR ensures that the entire channel is capable of properly indicating neutron flux. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. Internal test circuitry is used to feed pre-adjusted test signals into the preamplifier to verify channel alignment. It is not necessary to test the detector, because generating a meaningful test signal is difficult; the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output. This Frequency is the same as that employed for the same channels in the other applicable MODEs.

SR 3.3.12.3

Surveillance Requirement 3.3.12.3 is the performance of a CHANNEL CALIBRATION. ~~A CHANNEL CALIBRATION is performed every 24 months.~~ The surveillance test is a complete check and readjustment of the wide range logarithmic neutron flux monitor channel from the preamplifier input through to the remote indicators. The surveillance test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive surveillance tests. CHANNEL CALIBRATIONS must be performed consistent with the plant-specific setpoint analysis.

This SR is modified by a Note to indicate that it is not necessary to test the detector because generating a meaningful test signal is difficult; the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output. ~~This Frequency is the same as that employed for the same channels in the other applicable MODEs.~~

Insert 3

BASES

a violation of this LCO occur, the operator should check whether or not an SL may have been exceeded.

ACTIONS

A.1

Pressurizer pressure and RCS cold leg temperature are controllable and measurable parameters. Reactor Coolant System flow rate is not a controllable parameter and is not expected to vary during steady-state operation. With any parameter not within its LCO limit, action must be taken to restore the parameter.

The two hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause of the off normal condition, and to restore the readings within limits. The Completion Time is based on plant operating experience that shows the parameter can be restored in this time period.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within six hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds.

Six hours is a reasonable time that permits the plant power to be reduced at an orderly rate in conjunction with even control of steam generator (SG) heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Insert 3

~~Since Required Action A.1 allows a Completion Time of two hours to restore parameters that are not within limits, the 12 hour SR Frequency for pressurizer pressure is sufficient to ensure that the pressure can be restored to a normal operation, steady-state condition following load changes, and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and verify operation is within safety analysis assumptions.~~

BASES

SR 3.4.1.2

~~Since Required Action A.1 allows a Completion Time of two hours to restore parameters that are not within limits, the 12 hour SR Frequency for cold leg temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady state condition following load changes, and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.~~ Average Reactor Protective System (RPS) cold leg indication may be used for this SR as described in plant procedures. Use of the maximum RPS cold leg indication for this SR is acceptable and conservative at all times.

Insert 3

SR 3.4.1.3

~~The 12 hour SR Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour Frequency has been shown by operating experience to be sufficient to assess for potential degradation and to verify operation is within safety analysis assumptions.~~

SR 3.4.1.4

Measurement of RCS total flow rate is performed ~~once every 24 months~~. This verifies that the actual RCS flow rate is within the bounds of the analyses.

~~The Frequency of 24 months reflects the importance of verifying flow after a refueling outage where the core has been altered, which may have caused an alteration of flow resistance.~~

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR), Section 14.1.2, "Plant Characteristics Considered in Safety Analysis"

BASES

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

Reference 2, Section XI, Appendix E, may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The Completion Time of prior to entering MODE 4 forces the evaluation prior to entering a MODE where temperature and pressure can be significantly increased. The evaluation for a mild violation is possible within several days, but more severe violations may require special, event specific stress analyses or inspections.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within limits is required ~~every 30 minutes~~ when RCS P/T conditions are undergoing planned changes. ~~This Frequency is considered reasonable in view of the Control Room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.~~

Insert 3

The SR for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

BASES

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs, and minimizes the possibility of violating DNB limits. It should be noted that the reactor will trip and place the plant in MODE 3 as soon as the RPS senses less than 370,000 gpm** RCS flow.

The Completion Time of six hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification ~~every 12 hours~~ of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which help to ensure that forced flow is providing heat removal while maintaining the margin to DNB. ~~The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, Control Room indication and alarms will normally indicate loop status.~~

Insert 3

REFERENCES

1. UFSAR, Chapter 14, "Safety Analysis"

** ~~The Reactor Coolant Sy~~stem Flow Rate limit shall be ≥ 340,000 gpm through Unit 2, Cycle 14.

into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. When water is added without forced circulation, unmixed coolant could be introduced to the core, however water added with a boron concentration meeting the minimum SDM maintains an acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification ~~every 12 hours~~ that the required number of RCS loops are in operation. Verification includes flow rate, temperature, and pump status monitoring, which help ensure that forced flow is providing heat removal. ~~The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, Control Room indication and alarms will normally indicate loop status~~.

Insert 3

SR 3.4.5.2

This SR requires verification ~~every 12 hours~~ that the secondary side water level in each SG is > -50 inches. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. ~~The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within the safety analyses assumptions.~~

SR 3.4.5.3

Verification that the required number of RCPs are OPERABLE ensures that the single failure criterion is met and that an additional RCS loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs. ~~The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

C.1 and C.2

If no RCS or SDC loops are OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1, must be suspended and action to restore one RCS or SDC loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. When water is added without forced circulation, unmixed coolant could be introduced to the core, however water added with a boron concentration meeting the minimum SDM maintains an acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of decay heat removal. The action to restore must continue until one loop is restored to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification ~~every 12 hours~~ that one required loop is in operation. This ensures forced flow is providing heat removal. Verification includes flow rate, temperature, or pump status monitoring. T~~he 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, Control Room indication and alarms will normally indicate loop status~~.

Insert 3

SR 3.4.6.2

This SR requires verification ~~every 12 hours~~ of secondary side water level in the required SG(s) > -50 inches. An adequate SG water level is required in order to have a heat sink for removal of the core decay heat from the reactor coolant. ~~The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions~~.

BASES

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or SDC loop can be placed in operation, if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required loop components that are not in operation. For an RCS loop, the required component is a pump. For an SDC loop, the required components are the pump and valves. ~~The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

Insert 3

REFERENCES None

safe operation. When water is added without forced circulation, unmixed coolant could be introduced to the core, however water added with a boron concentration meeting the minimum SDM maintains an acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification ~~every 12 hours~~ that one SDC loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal. ~~The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation is within safety analyses assumptions. In addition, Control Room indication and alarms will normally indicate loop status.~~

Insert 3

The SDC flow is established to ensure that core outlet temperature is maintained sufficiently below saturation to allow time for swapover to the standby SDC loop should the operating loop be lost.

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ -50 inches ensures that redundant heat removal paths are available if the second SDC loop is inoperable. This surveillance test is required to be performed when the LCO requirement is being met by use of the SGs. If both SDC loops are OPERABLE, this SR is not needed. ~~The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions.~~

SR 3.4.7.3

Verification that the second SDC loop is OPERABLE ensures that redundant paths for decay heat removal are available. The requirement also ensures that the additional loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is

BASES

performed by verifying proper breaker alignment and power available to the required pumps and valves that are not in operation. This surveillance test is required to be performed when the LCO requirement is being met by one of two SDC loops, e.g., both SGs have < -50 inches water level. Insert 3 ~~The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience~~.

REFERENCES None

BASES

status and place it in operation must be initiated immediately. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. When water is added without forced circulation, unmixed coolant could be introduced to the core, however water added with a boron concentration meeting the minimum SDM maintains an acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification ~~every 12 hours~~ that one SDC loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal. ~~The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation and verify operation is within safety analyses assumptions.~~

Insert 3

SR 3.4.8.2

Verification that the required number of loops are OPERABLE ensures that redundant paths for heat removal are available and that additional loops can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and indicated power available to the required pumps and valves that are not in operation. ~~The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

REFERENCES

None

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR ensures that during steady-state operation, pressurizer water level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The surveillance test is performed by observing the indicated level. ~~The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications~~.

Insert 3

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) ~~The Frequency of 24 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable~~.

REFERENCES

1. NUREG-0737, II.E.3.1, "Clarification of TMI Action Plan Requirements," November 1980

of one hour or place the associated PORVs in override closed and restore at least one block valve to OPERABLE status within 72 hours, and the remaining block valve in five days, per Required Action C.2. The Completion Time of one hour to either restore the block valves or place the associated PORVs in override closed is reasonable based on the small potential for challenges to the system during this time and provides the operator time to correct the situation.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, then the plant must be brought to a MODE in which the LCO does not apply. The plant must be brought to at least MODE 3 within 6 hours and reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) within 12 hours. The Completion Time of six hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging safety systems. Similarly, the Completion Time of 12 hours to reduce any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) is reasonable considering that a plant can cool down within that time frame. In MODE 3 with any RCS cold leg temperature ≤ 365°F (Unit 1), ≤ 301°F (Unit 2) and in MODEs 4, 5, and 6, maintaining PORV OPERABILITY is required per LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

A CHANNEL FUNCTIONAL TEST is performed on each PORV instrument channel ~~every 92 days~~ to ensure the entire channel will perform its intended function when needed.

Insert 3

SR 3.4.11.2

Block valve cycling verifies that it can be closed if necessary. ~~The basis for the Frequency of 92 days is found in Reference 3.~~ If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance because opening the block valve is necessary to permit the PORV to be used for manual control of RCS pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is 120 hours, which is well within the allowable limits (25%) to extend the block valve surveillance interval ~~of 92 days~~. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Action fulfills the SR).

The Note modifies this SR by stating that this SR is not required to be performed with the block valve closed in accordance with the Required Actions of this LCO.

SR 3.4.11.3

Surveillance Requirement 3.4.11.3 requires complete cycling of each PORV. Power-operated relief valve cycling demonstrates its function. ~~The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice~~.

Insert 3

SR 3.4.11.4

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required ~~every 24 months~~ to adjust the whole channel so that it responds, and the valve opens within the required range and with accuracy to known input.

~~The 24 month Frequency considers operating experience with equipment reliability and matches the refueling outage Frequency.~~

REFERENCES

1. NUREG-0737, Paragraph II, G.I, "Clarification of TMI Action Plan Requirements," November 1980
2. Inspection and Enforcement Bulletin 79-05B, "Nuclear Incident at Three Mile Island - Supplement," April 21, 1979
3. ASME Code for Operation and Maintenance of Nuclear Power Plants

transient reasonable during the applicable MODEs. This action protects the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time of 48 hours to depressurize and vent the RCS is based on the time required to place the plant in this condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, verification that a maximum of one HPSI pump is only capable of manually injecting into the RCS, and automatic alignment of the HPSI loop MOVs, is prevented (by disabling the automatic opening features of the HPSI loop MOVs) is required. The HPSI pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control or by verifying their discharge valves are locked shut.

~~The 12 hour interval considers operating practice to regularly assess potential degradation and to verify operation within the safety analysis~~.

Insert 3

SR 3.4.12.3

Surveillance Requirement 3.4.12.3 requires verifying that the required RCS vent is open. ~~once every 12 hours for a valve that is unlocked open, and once every 31 days for a valve that is locked open~~.

The passive vent arrangement must only be open to be OPERABLE. This SR need only be performed if the vent is being used to satisfy the requirements of this LCO. The ~~Frequencies consider operating experience with mispositioning of unlocked and locked vent valves, respectively~~.

SR 3.4.12.4

The PORV block valve must be verified open ~~every 72 hours~~ to provide the flow path for each required PORV to perform its function when actuated. The valve can be remotely verified open in the main Control Room.

The block valve is a remotely controlled MOV. The power to the valve motor operator is not required to be removed, and the manual actuator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure event.

Insert 3

~~The 72 hour Frequency considers operating experience with accidental movement of valves having remote control and position indication capabilities available where easily monitored. These considerations include the administrative controls over main Control Room access and equipment control~~.

SR 3.4.12.5

Performance of a CHANNEL FUNCTIONAL TEST is required ~~every 31 days~~ to verify and, as necessary, adjust the PORV open setpoints. The CHANNEL FUNCTIONAL TEST will verify on a monthly basis that the PORV lift setpoints are within the LCO limit. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. Power-operated relief valve actuation could depressurize the RCS and is not required. ~~The 31 day Frequency considers experience with equipment reliability~~.

A Note has been added indicating this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to ≤ 365°F (Unit 1), ≤ 301°F (Unit 2). The test cannot be performed until the RCS is in the LTOP MODEs when the PORV lift setpoint can be reduced to the LTOP setting. The test

BASES

must be performed within 12 hours after entering the LTOP MODEs.

SR 3.4.12.6

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required ~~every 24 months~~ to adjust the whole channel so that it responds and the valve opens within the required LTOP range and with accuracy to known input.

Insert 3

~~The 24 month Frequency considers operating experience with equipment reliability and matches the typical refueling outage schedule.~~

REFERENCES

1. 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities
2. Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations, July 12, 1988
3. UFSAR, Section 4.2.2, Low Temperature Overpressure Protection
4. Generic Letter 90-06, Resolution of Generic Issues 70, "PORV and Block Valve Reliability," and 94, "Additional LTOP Protection for PWRs," June 25, 1990

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady-state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, and makeup and letdown). The surveillance is modified by two Notes. Note 1 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 water inventory balance; 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 leakoff 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. These leakage detection systems are specified in LCO 3.4.14.

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 100 gpd cannot be measured accurately by an RCS water inventory balance.

Insert 3

~~The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents~~.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 100 gpd through any one SG. Satisfying the

BASES

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 3.4.18, "Steam Generator Tube Integrity," should be evaluated. The 100 gpd limit is measured at hot plant conditions as described in Reference 4. 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, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady-state operation. For the RCS primary to secondary LEAKAGE determination, steady-state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, and makeup and letdown.

Insert 3

~~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 guidelines in Reference 4~~.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973
3. Nuclear Energy Institute (NEI) 97-06, Steam Generator Program Guidelines
4. Electric Power Research Institute, Pressurized Water Reactor Primary-to-Secondary Leakage Guidelines
5. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

the plant will not be operated in a degraded configuration for a lengthy time period.

D.1 and D.2

If any required Action of Conditions A, B, or C cannot be met within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If all required alarms and monitors are inoperable, no automatic means of monitoring leakage are available, an immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Surveillance Requirement 3.4.14.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence the channel is operating properly. ~~The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.~~

Insert 3

SR 3.4.14.2

Surveillance Requirement 3.4.14.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitors. 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. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by

BASES

other Technical Specification tests at least once per refueling interval with applicable extensions. ~~The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.~~

Insert 3

SR 3.4.14.3 and SR 3.4.14.4

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 Structure. ~~The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Operating experience has shown this Frequency is acceptable.~~

REFERENCES

1. UFSAR
2. Regulatory Guide 1.45, Revision 0, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973

BASES

C.1

With the gross activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.

The change within six hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of six hours is required to reach MODE 3 below 500°F from full power conditions and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

The SR requires performing a gamma isotopic analysis, as a measure of the gross activity of the reactor coolant~~, at least once per seven days~~. While $\bar{E}$ is basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this gamma isotopic measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This SR provides an indication of any increase in gross activity.

Trending the results of this SR allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The SR is applicable in MODEs 1 and 2, and in MODE 3 with RCS average temperature at least 500°F. ~~The seven day Frequency considers the unlikelihood of a gross fuel failure during the time.~~

Insert 3

SR 3.4.15.2

This SR is performed to ensure iodine remains within limits during normal operation and following fast power changes when fuel failure is more apt to occur. ~~The 14 day Frequency is adequate to trend changes in the iodine activity level considering gross activity is monitored every 7 days.~~ The Frequency, between two hours and six hours after a power change of ≥ 15% RTP within a one hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

The SR is modified by a Note which requires the surveillance test to only be performed in MODE 1. This is required because the level of fission products generated in other MODEs is much less. Also, fuel failures associated with fast power changes is more apt to occur in MODE 1 than in MODEs 2 and 3.

SR 3.4.15.3

A radiochemical analysis for $\bar{E}$ determination is required ~~every 184 days (six months)~~ with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. ~~The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.~~

Insert 3

This SR has been modified by a Note that indicates sampling is not required to be performed until 31 days after 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. This ensures the radioactive materials are at equilibrium so that analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

BASES

LCO

This LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2 (after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without this LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODEs 1 and 2), and the appropriate tests could not be performed.

In MODE 2, where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no flow conditions provided THERMAL POWER is < 5% RTP and the reactor trip setpoints of the OPERABLE power level channels are set ≤ 15% RTP. These limits ensure no SLs or fuel design limits will be violated.

The exception is allowed even though there are no bounding safety analyses. These tests are allowed since they are performed under close supervision during the test program and provide valuable information on the plant's capability to cool down without offsite power available to the RCPs.

APPLICABILITY

This LCO ensures that the plant will not be operated in MODE 1 without forced circulation. It only allows testing under these conditions while in MODE 2. This testing establishes that heat input from nuclear heat does not exceed the natural circulation heat removal capabilities. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.

ACTIONS

A.1

If THERMAL POWER increases to > 5% RTP, the reactor must be tripped immediately. This ensures the plant is not placed in an unanalyzed condition and prevents exceeding the specified acceptable fuel design limits.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

THERMAL POWER must be verified to be within limits ~~once per hour~~ to ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS~~.~~ ~~The hourly Frequency has been shown by operating practice to be sufficient to regularly assess conditions for potential~~

Insert 3

BASES

~~degradation and verify operation is within the LCO limits~~. ~~Plant operations are conducted slowly during the performance of PHYSICS TESTS, and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit~~.

SR 3.4.16.2

Within 12 hours of initiating startup or PHYSICS TESTS, a CHANNEL FUNCTIONAL TEST must be performed on each logarithmic power level neutron flux monitoring channel to verify OPERABILITY and adjust setpoints to proper values. This will ensure that the RPS is properly aligned to provide the required degree of core protection during startup or the performance of the PHYSICS TESTS. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. The interval is adequate to ensure that the appropriate equipment is OPERABLE prior to the tests to aid the monitoring and protection of the plant during these tests.

REFERENCES

1. 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, Section XI

BASES

APPLICABILITY

The LCO ensures that while within this LCO the plant will not be operated in any other MODE besides MODEs 4 and 5 without forced circulation. This is because the MODEs of Applicability for this Specification are MODEs 4 and 5. This Specification allows testing and maintenance to be performed on the SDC System while SDC is required to be OPERABLE.

ACTIONS

A.1

If one or more requirements of the LCO are not met, all activities being performed under this STE must be immediately suspended. These activities are local leak rate testing of the SDC penetration and maintenance on valves in the SDC System. The Completion Time to suspend these activities immediately ensures the plant is not placed in an unanalyzed condition and prevents exceeding the specified acceptable fuel design limits.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

Xenon reactivity must be verified to be within limits once within one hour prior to suspending the reactor coolant circulation requirements of LCO 3.4.6, LCO 3.4.7, and LCO 3.4.8. The frequency of once within one hour prior to suspending the applicable RCS Loops LCO will ensure that the xenon reactivity is within limits and trending toward stability prior to suspending forced flow cooling. This will ensure no SLs or fuel design limits will be violated while testing or maintenance are being conducted.

SR 3.4.17.2 and SR 3.4.17.3

Verifying the charging pumps are de-energized and the charging flow paths are isolated, ensures that the major source of a boron reduction is not available. These two SRs support the requirement that no source be available that could cause an RCS boron concentration reduction. ~~These SRs are required to be verified at a frequency of one hour. The one hour frequency is sufficient to ensure that these sources will not be available to cause a reduction of the RCS boron concentration.~~

Insert 3

BASES

Subsequent performance of these SRs after the initial verification that the charging pumps are de-energized and the charging flow paths are isolated, may be performed administratively.

SR 3.4.17.4

This SR requires that a SHUTDOWN MARGIN verification be performed in accordance with SR 3.1.1.1 ~~once per eight hours. The normal Frequency for these SRs is once per 24 hours. The eight hour Frequency reflects that no forced flow cooling is available and that the SHUTDOWN MARGIN should be verified more frequently. The eight hour Frequency is sufficient to ensure that the SHUTDOWN MARGIN remains within limits while under this STE~~.

Insert 3

REFERENCES	None

conditions in an orderly manner and without challenging plant systems.

D.1

If more than one SIT is inoperable, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification ~~every 12 hours~~ that each SIT isolation valve is fully open, as indicated in the Control Room, ensures that SITs are available for injection and ensures timely discovery if a valve should be partially closed. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor-operated valve should not change position with power removed, a closed valve could result in not meeting accident analysis assumptions. ~~A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure the unlikelihood of a mispositioned isolation valve~~.

Insert 3

SR 3.5.1.2 and SR 3.5.1.3

Safety injection tank borated water volume and nitrogen cover pressure should be verified to be within specified limits ~~every 12 hours~~ in order to ensure adequate injection during a LOCA. ~~Due to the static design of the SITs, a 12 hour Frequency usually allows the operator sufficient time to identify changes before the limits are reached~~. ~~Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends~~.

SR 3.5.1.4

Six months is reasonable for verification by sampling to determine that each SIT's boron concentration is within the required limits, because the static design of the SITs limits the ways in which the concentration can be changed. ~~This Frequency is adequate to identify changes that could occur from mechanisms, such as stratification or inleakage~~.

BASES

Verification consists of monitoring inleakage or sampling. The inleakage is monitored ~~every 12 hours~~ by monitoring tank level. Sampling of each tank is done ~~every six months~~. All intentional sources of level increase are maintained administratively to ensure SIT boron concentrations are within technical specification limits. The boron concentration of each tank is verified prior to startup from outages. A sample of the SIT is required, to verify boron concentration, if 10 inches or greater of inleakage has occurred since last sampled.

Sampling the affected SIT (by taking the sample at the discharge of the operating HPSI pump) within one hour prior to a 1% volume increase of normal tank volume, will ensure the boron concentration of the fluid to be added to the SIT is within the required limit prior to adding inventory to the SIT(s).

SR 3.5.1.5

Verification ~~every 31 days~~ that power is removed from each SIT isolation valve operator, by maintaining the feeder breaker open under administrative control, when the pressurizer pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an SIT motor-operated isolation valve. If this were to occur, only two SITs would be available for injection, given a single failure coincident with a LOCA. ~~Since installation and removal of power to the SIT isolation valve operators is conducted under administrative control, the 31 day Frequency was chosen to provide additional assurance that power is removed~~.

Insert 3

This SR allows power to be supplied to the motor-operated isolation valves when RCS pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during unit startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves. Should closure of a valve occur in spite of the interlock, the safety injection signal provided to the valves would open a closed valve in the event of a LOCA.

BASES

An event accompanied by a loss of offsite power and the failure of an emergency diesel generator can disable one ECCS train until power is restored. A reliability analysis (Reference 3) has shown that the impact with one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 4 describes situations in which one component, such as a SDC total flow control valve, can disable both ECCS trains. With one or more components inoperable, such that 100% of the equivalent flow to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < 1750 psia within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. MOV-659 and MOV-660 are secured in position by interrupting the control signal to the valve operator via a key switch in the Control Room. Power is removed from the valve operator for CV-306 by isolating the air supply to the valve positioner. These actions ensure that the valves cannot be inadvertently misaligned. These valves are of the type described in Reference 4, which can disable the function of both ECCS trains and invalidate the accident analysis. ~~A 12 hour Frequency is considered reasonable in view of other administrative controls ensuring that a mispositioned valve is an unlikely possibility.~~

Insert 3

SR 3.5.2.2

Verifying the correct alignment for manual, power-operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve automatically repositions within the proper stroke time. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position.

Insert 3

~~The 31 day Frequency is appropriate because the valves are operated under procedural control and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.~~

SR 3.5.2.3

Periodic surveillance testing of the HPSI and LPSI pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the American Society of Mechanical Engineers Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the unit safety analysis. Surveillance Requirements are specified in the Inservice Testing Program, which encompasses American Society of Mechanical Engineers Code. American Society of Mechanical Engineers Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.4

The Surveillance Requirement was deleted in Amendment Nos. 260/237.

SR 3.5.2.5, SR 3.5.2.6, and SR 3.5.2.7

These SRs demonstrate that each automatic ECCS valve actuates to the required position on an actual, or simulated SIAS, and on a recirculation actuation signal; that each ECCS pump starts on receipt of an actual or simulated SIAS; and that the LPSI pumps stop on receipt of an actual or simulated recirculation actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. In order to assure the results of the low temperature overpressure protection analysis remain bounding, whenever flow testing into the RCS is required at RCS temperatures ≤ 365°F (Unit 1), ≤ 301°F (Unit 2), the HPSI pump shall recirculate RCS water (suction from the RWT isolated) or the requirements of LCO 3.4.12, shall be satisfied. ~~The 24 month Frequency is based on the need to perform these surveillance tests under the conditions that apply during a plant outage and the potential for unplanned transients if the surveillance tests were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment~~. The actuation logic is tested as part of the Engineered Safety Feature Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

Insert 3

SR 3.5.2.8

Periodic inspection of the containment sump ensures that it is unrestricted and stays in proper operating condition. ~~The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during an outage, on the need to have access to the location, and on the potential for unplanned transients if the surveillance test were performed with the reactor at power. This Frequency is sufficient to detect abnormal degradation and is confirmed by operating experience.~~

SR 3.5.2.9

Verifying that the SDC System open-permissive interlock is OPERABLE ensures that the SDC suction isolation valves are prevented from being remotely opened when RCS pressure, is at or above, the SDC System design suction pressure of

BASES

350 psia. The suction piping of the LPSI pumps, is the SDC component with the limiting design pressure rating. The interlock provides assurance that double isolation of the SDC System from the RCS is preserved whenever RCS pressure, is at or above, the design pressure. The 309 psia value specified in the Surveillance is the actual pressurizer pressure at the instrument tap elevation for PT-103 and PT-103-1 when the SDC System suction pressure is 350 psia. The procedure for this surveillance test contains the required compensation to be applied to this value to account for instrument uncertainties. This surveillance test is normally performed using a simulated RCS pressure input signal. ~~The 24 month Frequency is based on the need to perform this surveillance test under conditions that apply during an outage. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) for the equipment.~~

Insert 3

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"
3. Nuclear Regulatory Commission Memorandum to V. Stello, Jr., from R. L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975
4. Inspection and Enforcement Information Notice No. 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs," January 6, 1987

restore the RWT to OPERABLE status is based on this condition simultaneously affecting multiple redundant trains.

C.1 and C.2

If the RWT cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1 and SR 3.5.4.2

Refueling water tank borated water temperature shall be verified ~~every 24 hours~~ to be within the limits assumed in the accident analysis. ~~This Frequency has been shown to be sufficient to identify temperature changes that approach either acceptable limit~~.

Insert 3

The SRs are modified by a Note that eliminates the requirement to perform this surveillance test when ambient air temperatures are within the operating temperature limits of the RWT. With ambient temperatures within this range, the RWT temperature should not exceed the limits.

Surveillance Requirement 3.5.4.2 is modified by an additional Note which requires the SR to be met in MODE 1 only. A SR is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a SR, even without a surveillance test specifically being "performed," constitutes a SR not "met." This reflects the maximum coolant temperature assumptions in the LOCA analysis.

SR 3.5.4.3

Above minimum RWT water volume level shall be verified ~~every seven days~~. This Frequency ensures that a sufficient initial water supply is available for injection and to support continued ESF pump operation on recirculation.

BASES

~~Since the RWT volume is normally stable and is provided with a Low Level Alarm, a seven day Frequency is appropriate and has been shown to be acceptable through operating experience.~~

SR 3.5.4.4

Boron concentration of the RWT shall be verified ~~every seven days~~ to be within the required range. This Frequency ensures that the reactor will remain subcritical following a LOCA. Further, it ensures that the resulting sump pH will be maintained in an acceptable range such that boron precipitation in the core will not occur earlier than predicted, and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. ~~Since the RWT volume is normally stable, a seven day sampling Frequency is appropriate and has been shown through operating experience to be acceptable.~~

Insert 3

REFERENCES

1. UFSAR, Chapters 6, "Engineered Safety Features," and 14, "Safety Analysis"

BASES

brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Periodic determination of the equivalent weight of STB in Containment must be performed due to the possibility of leaking valves and components in the Containment Building that could cause dissolution of the STB during normal operation. A ~~Frequency of 24 months~~ verification is required to determine visually that a minimum of 13,750 lbm is contained in the STB baskets. This requirement ensures that there is an adequate mass of STB to adjust the pH of the post-LOCA sump solution to a value ≥ 7.0.

Insert 3

~~The periodic verification is required every 24 months, since access to the STB baskets is only feasible during outages, and normal fuel cycles are scheduled for 24 months. Operating experience has shown this SR Frequency acceptable, due to the margin in the weight of STB placed in the Containment Building.~~

SR 3.5.5.2

Testing must be performed to ensure the solubility and buffering ability of the STB after exposure to the containment environment. A representative sample of 2.74 ± 0.05 grams of STB, from one of the baskets in Containment is submerged in 1.0 ± 0.01 liters of water at a boron concentration of 3074 ± 50 ppm, and at the standard temperature of 120 ± 5°F. Within four hours without agitation, the solution is decanted and mixed, the temperature adjusted to 77 ± 2°F, and the pH measured. The solution pH should be ≥ 7.0. The representative sample weight is based on the minimum required STB equivalent weight of 13,750 lbm, and maximum possible post-LOCA sump water mass of 4,608,356 lbm, normalized to buffer a 1.0 ± 0.01 liter sample. The boron concentration of the

BASES

test water is representative of the maximum possible boron concentration corresponding to the maximum possible post-LOCA sump volume. Agitation of the test solution is prohibited, since an adequate standard for the agitation intensity cannot be specified. A test time of four hours would allow time for the dissolved STB to naturally diffuse through the sample solution. A test time of less than four hours is more conservative than a test time of longer than four hours because the longer time could permit additional STB to dissolve, if excess STB was available. In the post-LOCA containment sump, rapid mixing would occur, significantly decreasing the actual amount of time before the required pH is achieved. This would ensure compliance with the Standard Review Plan requirement of a pH ≥ 7.0 by the onset of recirculation after a LOCA.

Insert 3

REFERENCES None

BASES

designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the air lock is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. ~~The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a plant outage and the potential for loss of the Containment Structure OPERABILITY if the surveillance test was performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during use of the air lock~~.

Insert 3

REFERENCES 1. UFSAR

Required Action C.2 is modified by a Note that applies to valves and blind flanges, located in high radiation areas, and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1 and D.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

This SR ensures that the containment vent valves are closed as required, or, if open, open for an allowable reason. If a containment vent valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the containment vent valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for surveillance tests that require the valves to be open. The containment vent valves are capable of closing in the environment, following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. ~~The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.2.~~

Insert 3

SR 3.6.3.2

This SR requires verification that each containment isolation manual valve and blind flange located outside the Containment Structure, and not locked, sealed, or otherwise

BASES

secured, and required to be closed during accident conditions is closed. The SR helps to ensure that post-accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside the Containment Structure and capable of being mispositioned are in the correct position. ~~Since verification of valve position for containment isolation valves outside the Containment Structure is relatively easy, the 31 day Frequency is based on engineering judgment, and was chosen to provide added assurance of the correct positions.~~ Insert 3 Containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODEs 1, 2, 3, 4 and for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

<u>SR 3.6.3.3</u>

This SR requires verification that each containment isolation manual valve and blind flange located inside the Containment Structure, and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed. The SR helps to ensure that post-accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For containment isolation valves inside the Containment Structure, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time that they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODEs 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test, ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program. The isolation time limits are contained in Reference 2.

SR 3.6.3.5

Automatic containment isolation valves close on an isolation signal [containment isolation signal Channels A or B, or safety injection actuation signal (SIAS) Channels A or B] to prevent leakage of radioactive material from the Containment Structure following a DBA. This SR ensures each automatic containment isolation valve will actuate to its isolation position on a containment isolation actuation signal. This surveillance test is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 24 month Frequency was developed considering it is prudent that this SR be performed only during a unit outage, since isolation of penetrations would eliminate cooling water flow and~~

Insert 3

BASES

~~disrupt normal operation of many critical components~~. ~~Operating experience has shown that these components usually pass this SR when performed on the 24 month Frequency~~. ~~Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

REFERENCES

1. UFSAR, Chapter 5, "Structures", Figure 5-10
2. UFSAR, Chapter 5, "Structures", Table 5-3

BASES

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner, and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that operation remains within the limits assumed in the accident analysis. ~~The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of containment pressure variations during the applicable MODEs. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the Control Room, including alarms, to alert the operator to an abnormal containment pressure condition~~.

Insert 3

REFERENCES

None

BASES

within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken from the containment dome [1(2)-TI-5309] and the containment reactor cavity [1(2)-TI-5311] temperature indicators selected to provide a representative sample of the overall containment atmosphere. ~~The 24 hour Frequency of this SR is considered acceptable based on the observed slow rates of temperature increase within the Containment Structure as a result of environmental heat sources (due to the large volume of the Containment Structure). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the Control Room, including alarms, to alert the operator to an abnormal containment temperature condition.~~

Insert 3

REFERENCES

1. UFSAR, Section 14.20, "Containment Response"

E.1 and E.2

If the Required Actions and associated Completion Times of Conditions C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With two containment spray trains or any combination of three or more Containment Spray and Cooling Systems trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verifying, through a system walkdown, that those valves outside the Containment Structure and capable of potentially being mispositioned are in the correct position.

Insert 3

SR 3.6.6.2

Starting each containment cooling train fan unit from the Control Room and operating it for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected and corrective action taken. ~~The 31 day Frequency of this SR~~

BASES

~~was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of a significant degradation of the containment cooling train occurring between surveillances and has been shown to be acceptable through operating experience~~.

SR 3.6.6.3

of 31 days

Verifying a service water flow rate of ≥ 2000 gpm to each cooling unit when the full flow service water outlet valves are fully open provides assurance that the design flow rate assumed in the safety analyses will be achieved (Reference 1, Chapter 7). Also considered in selecting this Frequency were the known reliability of the Service Water System, the two train redundancy, and the low probability of a significant degradation of flow occurring between surveillance tests.

SR 3.6.6.4

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Reference 2. Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.5 and SR 3.6.6.6

These SRs verify that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation signal (i.e., the appropriate Engineered Safety Feature Actuation System signal). This SR is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The

BASES

~~24 month Frequency is based on the need to perform these surveillance tests under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance tests were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance tests when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

Insert 3

The surveillance test of containment sump isolation valves is also required by SR 3.5.2.5. A single surveillance test may be used to satisfy both requirements.

SR 3.6.6.7

This SR verifies that each containment cooling train actuates upon receipt of an actual or simulated actuation signal (i.e., the appropriate Engineered Safety Feature Actuation System signal). ~~The 24 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 24 month Frequency.~~

SR 3.6.6.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through check valve bonnets. Performance of this SR demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the Containment Structure during an accident is not degraded. Due to the passive design of the nozzle, a test after maintenance that could result in nozzle blockage is considered adequate. Maintenance that could result in nozzle blockage is generally loss of foreign material control or a flow of borated water through a nozzle. Should either of these events occur, a supervisory evaluation will be required to determine whether nozzle blockage is a possible result of the event.

b. The fact that, even with no IRS train in operation, almost the same amount of iodine would be removed from the containment atmosphere through absorption by the Containment Spray System; and

c. The fact that the Completion Time is adequate to make most repairs.

B.1

If two IRS trains are inoperable, one must be restored to OPERABLE status within one hour. The one hour Completion Time allows the swing train to be aligned to the appropriate bus to ensure each of the two remaining trains are powered from separate and independent buses. The one hour, also allows time to restore one train to OPERABLE status prior to initiating a plant shutdown. This is reasonable considering that a plant shutdown is a plant transient.

C.1 and C.2

If the IRS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Initiating each IRS train from the Control Room and operating it for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that motor failure can be detected for corrective action. ~~The 31 day Frequency was developed considering the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System independent of the IRS.~~

Insert 3

SR 3.6.8.2

This SR verifies that the required IRS filter testing is performed in accordance with the Ventilation Filter Testing Program. The IRS filter tests are in accordance with portions of Reference 2. The Ventilation Filter Testing Program includes testing high efficiency particulate air filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the Ventilation Filter Testing Program.

SR 3.6.8.3

The automatic startup test verifies that both trains of equipment start upon receipt of an actual or simulated test signal (Engineered Safety Feature Actuation System). ~~The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the Frequency was developed considering that the system equipment OPERABILITY is demonstrated on a 31 day Frequency by SR 3.6.8.1.~~

Insert 3

REFERENCES

1. UFSAR
2. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978

prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulations; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position.

Insert 3

~~The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions~~.

SR 3.7.3.2

Cycling each testable, remote-operated valve that is not in its operating position, provides assurance that the valves will perform as required. Operating position is the position that the valve is in during normal plant operation. This is accomplished by cycling each valve at least one cycle. This SR ensures that valves required to function during certain scenarios, will be capable of being properly positioned. The Frequency is based on engineering judgment that when cycled in accordance with the Inservice Testing Program, these valves can be placed in the desired position when required.

SR 3.7.3.3

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ($\geq$ 2800 ft for the steam-driven pump and $\geq$ 3100 ft for the motor-driven pump), ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by Reference 2. Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, discussed in Reference 2, at three month intervals satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred up to 24 hours until suitable test conditions are established. This deferral is required because there is an insufficient steam pressure to perform the test.

SR 3.7.3.4

This SR ensures that AFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an AFAS signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal (verification of flow-modulating characteristics is not required). This SR is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. The 24 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.~~

Insert 3

This SR is modified by a Note indicating that the SR should be deferred up to 24 hours until suitable test conditions have been established.

SR 3.7.3.5

This SR ensures that the AFW pumps will start in the event of any accident or transient that generates an AFAS signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. ~~The 24 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.~~

This SR is modified by a Note. The Note indicates that the SR should be deferred up to 24 hours until suitable test conditions are established.

SR 3.7.3.6

This SR ensures that the AFW system is capable of providing a minimum nominal flow to each flow leg. This ensures that the minimum required flow is capable of feeding each flow leg. The test may be performed on one flow leg at a time. The SR is modified by a Note which states, the SR is not required to be performed for the AFW train with the turbine-driven AFW pump until up to 24 hours after reaching 800 psig in the steam generators. The Note ensures that proper test conditions exist prior to performing the test using the turbine-driven AFW pumps. ~~The 24 month Frequency coincides with performing the test during refueling outages~~. Insert 3

SR 3.7.3.7

This SR ensures that the AFW System is properly aligned by verifying the flow path to each steam generator prior to entering MODE 2 operation, after 30 days in MODEs 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, and other administrative controls to ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, the OPERABILITY of the flow paths is verified following extended outages to determine that no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned. Minimum nominal flow to each flow leg is ensured by performance of SR 3.7.3.6.

REFERENCES

1. UFSAR, Section 10.3
2. ASME Code for Operation and Maintenance of Nuclear Power Plants

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the affected unit(s) must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit(s) must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the CST contains the required usable volume of cooling water. (This volume ≥ 150,000 gallons per unit in the MODE of Applicability.) ~~The 12 hour Frequency is based on operating experience, and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the Control Room, including alarms, to alert the operator to abnormal CST volume deviations~~.

Insert 3

Although the volume in the CST for each unit is required to be 150,000 gallons, the total combined volume for both units is 300,000 gallons.

REFERENCES

1. UFSAR

and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CC loop cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the CC flow path provides assurance that the proper flow paths exist for CC operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

This SR is modified by a Note indicating that the isolation of the CC components or systems may render those components inoperable but does not affect the OPERABILITY of the CC System.

Insert 3

~~The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~

SR 3.7.5.2

This SR verifies proper automatic operation of the CC valves on an actual or simulated safety injection actuation signal

(SIAS). The CC System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This SR is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

Insert 3

SR 3.7.5.3

This SR verifies proper automatic operation of the CC pumps on an actual or simulated SIAS. The CC System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. ~~The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

REFERENCES 1. UFSAR

achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the SRW flow path ensures that the proper flow paths exist for SRW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR is modified by a Note indicating that the isolation of the SRW components or systems may render those components inoperable but does not affect the OPERABILITY of the SRW System.

Insert 3

~~The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~

SR 3.7.6.2

This SR verifies proper automatic operation of the SRW System valves on an actual or simulated actuation signal (SIAS or CSAS). The SRW System is a normally operating system that cannot be fully actuated as part of normal testing. This surveillance test is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage, and the potential for an unplanned transient if the surveillance test were performed with the reactor at~~

BASES

~~power. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

SR 3.7.6.3

The SR verifies proper automatic operation of the SRW System pumps on an actual or simulated actuation signal (SIAS or CSAS). The SRW System is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. ~~Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

Insert 3

REFERENCES 1. UFSAR

BASES

~~The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions~~.

Insert 3

SR 3.7.7.2

This SR verifies proper automatic operation of the SW System valves on an actual or simulated actuation signal (SIAS). The SW System is a normally operating system that cannot be fully actuated as part of the normal testing. This surveillance test is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 24 month Frequency is based on the need to perform this surveillance test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power~~. ~~Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ Note: There are currently no SW valves with an Engineered Safety Feature Actuation System signal since automatic system reconfiguration during a LOCA is not required.

SR 3.7.7.3

The SR verifies proper automatic operation of the SW System pumps on an actual or simulated actuation signal (SIAS). The SW System is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. ~~Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

REFERENCES

1. UFSAR, Section 9.5.2.3, "Saltwater System"

normal operating conditions on this system are not severe, testing each required CREVS filter train once every month provides an adequate check on this system.

~~The 31 day Frequency is based on the known reliability of the equipment, and the two filter train redundancy available~~.

Insert 3

SR 3.7.8.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREVS filter tests are in accordance with portions of Reference 2. The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.8.3

This SR verifies each CREVS train starts and operates on an actual or simulated actuation signal (CRRS). ~~This test is conducted on a 24 month Frequency. This Frequency is adequate to ensure the CREVS is capable of starting and operating on an actual or simulated CRRS~~.

SR 3.7.8.4

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 the CRE occupants calculated in the licensing basis analysis of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analysis of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition E must be entered. Options for restoring the CRE boundary to OPERABLE status include changing the

BASES

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met in MODEs 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1

If both CRETS trains are inoperable in MODEs 1, 2, 3, or 4, or during movement of irradiated fuel assemblies, the CRETS may not be capable of performing the intended function and the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately and movement of irradiated fuel must be suspended immediately. This does not preclude the movement of fuel assemblies to a safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies each required CRETS train has the capability to maintain Control Room temperature ≤ 104°F for ≥ 12 hours in the recirculation mode. During this test, the backup Control Room air conditioner is to be de-energized. This SR consists of a combination of testing. ~~A 24 month Frequency is appropriate, since significant degradation of the CRETS is slow and is not expected over this time period.~~

Insert 3

REFERENCES

1. UFSAR, Section 9.8.2.3, "Auxiliary Building Ventilating Systems"

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

The SR requires verification ~~every 12 hours~~ that the SFPEVS is in operation. Verification includes verifying that one exhaust fan is operating and discharging into the ventilation stack. ~~The Frequency of 12 hours is sufficient considering that the operators will be focused on the movement of recently irradiated fuel assemblies within the Auxiliary Building. Thus, if anything were to occur to cause cessation of operation of the SFPEVS, it would be quickly identified.~~

Insert 3

SR 3.7.11.2

Deleted.

SR 3.7.11.3

This SR verifies the integrity of the spent fuel storage pool area. The ability of the spent fuel storage pool area to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the SFPEVS. During operation, the spent fuel storage pool area is designed to maintain a slight negative pressure in the spent fuel storage pool area, with respect to adjacent areas, to ensure that exhausted air is directed to the ventilation stack.

~~This test is conducted on a 24 month Frequency. This Frequency is adequate to ensure the SFPEVS is capable of maintaining a negative pressure.~~

REFERENCES

1. UFSAR
2. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

during this time period, and the consideration that the remaining train can provide the required capability.

B.1 and B.2

If the inoperable train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

The test is performed by initiating the system from the Control Room, ensuring flow through the HEPA filter and charcoal adsorber train, and verifying this system operates for ≥ 15 minutes. ~~The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.~~

Insert 3

SR 3.7.12.2

This SR verifies the performance of PREVS filter testing in accordance with the VFTP. The PREVS filter tests are in accordance with portions of Reference 2. The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

BASES

SR 3.7.12.3

This SR verifies that each PREVS train starts and operates on an actual or simulated actuation signal (Containment Isolation Signal). ~~This test is conducted on a 24 month Frequency. This Frequency is adequate to ensure the PREVS is capable of starting and operating on an actual or simulated Containment Isolation Signal~~.

Insert 3

REFERENCES

1. UFSAR
2. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978
3. Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, June 2003
4. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

The SFP water level satisfies 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Reference 1, Section 14.18). As such, it is the minimum required for fuel storage, reconstitution, and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the SFP since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the SFP water level is lower than the required level, the movement of irradiated fuel assemblies in the SFP is immediately suspended. This effectively precludes a spent fuel handling accident from occurring. This does not preclude moving a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODEs 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODEs 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies sufficient SFP water is available in the event of a fuel handling accident. The water level in the SFP must be checked periodically. ~~The seven day Frequency is appropriate, because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.~~

Insert 3

BASES

secondary specific activity cannot be restored to within limits in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotope 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 3

REFERENCES

1. UFSAR, Chapter 14, "Safety Analysis"
2. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

BASES

ACTIONS

A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the concentration of boron in the SFPs is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limits.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies that the concentration of boron in the SFPs is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. ~~The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.~~ Insert 3

REFERENCES

None

SR 3.8.1.1 and SR 3.8.1.2

These SRs assure 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 Frequency of once within one hour after substitution for a 500 kV circuit and ~~every eight hours~~ periodically thereafter, for SR 3.8.1.1 was established to ensure that the breaker alignment for the SMECO circuit (which does not have Control Room indication) is in its correct position although breaker position is unlikely to change. ~~The seven day Frequency for SR 3.8.1.2 is adequate since the 500 kV circuit 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 3

Surveillance Requirement 3.8.1.1 is modified by a Note which states that this SR is only required when SMECO is being credited for an offsite source. This SR will prevent unnecessary testing on an uncredited circuit.

SR 3.8.1.3 and SR 3.8.1.9

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.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 2 for SR 3.8.1.3) to indicate that all DG starts for these surveillance tests may be preceded by an engine prelube period and followed by a warmup period prior to loading by an engine prelube period.

For the purposes of SR 3.8.1.9 testing, the DGs are required to start from standby conditions only for SR 3.8.1.9. Standby conditions for a DG mean the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

In order to reduce stress and mechanical wear on diesel engines, the DG manufacturers recommend a modified start in which the starting speed of DGs is limited, warmup is limited to this lower speed, and the DGs are gradually accelerated to synchronous speed prior to loading. This is the intent of Note 3, which is only applicable when such modified start procedures are recommended by the manufacturer.

Surveillance Requirement 3.8.1.9 requires that ~~at a 184 day Frequency,~~ the DG starts from standby conditions and achieves required voltage and frequency within 10 seconds. The minimum voltage and frequency stated in the SR are those necessary to ensure the DG can accept DBA loading while maintaining acceptable voltage and frequency levels. The 10 second start requirement supports the assumptions of the design basis loss of coolant accident analysis in Reference 2, Chapter 14.

Since SR 3.8.1.9 requires a 10 second start, it is more restrictive than SR 3.8.1.3, and it may be performed in lieu of SR 3.8.1.3.

Insert 3

~~The 31 day Frequency for SR 3.8.1.3 is consistent with Reference 4 and Reference 3. The 184 day Frequency for SR 3.8.1.9 is a reduction in cold testing consistent with Reference 7. This Frequency provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.~~

SR 3.8.1.4

This SR verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to 4000 kW for No. 1A DG and greater than or equal to 90% of the continuous duty rating for the remaining DGs. The 90% minimum load limit is consistent with Reference 3 and is acceptable because testing of these DGs at post-accident load values is performed by SR 3.8.1.11. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

BASES

Although no power factor requirements are established by this SR, the DG 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 1.0 is an operational limitation. ~~The 31 day Frequency for this SR is consistent with Reference 3~~.

Insert 3

This SR is modified by four Notes. Note 1 indicates that the diesel engine runs for this surveillance test may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients because of changing bus loads do not invalidate this test. Note 3 indicates that this surveillance test shall be conducted on only one DG at a time in order to prevent routinely paralleling multiple DGs and to minimize the potential for effects from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.5

This SR 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 required by the SR is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of one hour of DG operation at full load plus 10%.

Insert 3

~~The 31 day Frequency is adequate to assure 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.~~

SR 3.8.1.6

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 from the fuel oil day tanks ~~once every 31 days~~ eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling.

In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from 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 SR Frequencies are consistent with Reference 8. This SR is for preventive 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 test.

Insert 3

SR 3.8.1.7

This SR demonstrates that one 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 SR 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 Frequency for this SR is 31 days. The 31-day Frequency corresponds to the design of the fuel transfer system. The design of fuel transfer systems is such that pumps will operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day tanks during or following DG testing. In such a case, a 31-day Frequency is appropriate.~~

SR 3.8.1.8

Under accident and loss of offsite power conditions loads are sequentially connected to the bus by the automatic load sequencer (this SR verifies steps 1 through 5). The sequencing logic controls the permissive and closing signals to breakers to prevent overloading of the DGs due to high motor starting currents. The 10% load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load, and that safety analysis assumptions regarding ESF equipment time delays are not violated. The UFSAR provides a summary of the automatic loading of ESF buses.

~~The Frequency of 31 days is consistent with DG monthly testing and is sufficient to ensure the load sequencer operation as required~~.

Insert 3

SR 3.8.1.9

See SR 3.8.1.3.

SR 3.8.1.10

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 24 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 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

SR 3.8.1.11

This SR provides verification that the DG can be operated at a load greater than predicted accident loads for at least 60 minutes ~~once per 24 months~~. Operation at the greater than calculated accident loads will clearly demonstrate the ability of the DGs to perform their safety function. In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, testing must be performed using a DG load greater than or equal to calculated accident load and using a power factor ≤ 0.84 for No. 1A DG and ≤ 0.83 for Nos. 1B, 2A, and 2B DGs. These power factors are chosen based on the calculated highest kW value of DG loads during the postulated design basis accidents.

In addition, the post-accident load for No. 1A DG is significantly lower than the continuous rating of No. 1A DG.

BASES

To ensure No. 1A DG performance is not degraded, routine monitoring of engine parameters should be performed during the performance of this SR for No. 1A DG (Reference 9).

This SR is modified by two Notes, Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit will not invalidate the test. Note 2 ensures that the DG is tested under load conditions that are as close to design basis conditions as practicable. When synchronized with offsite power, testing should be performed at a power factor of ≤ 0.84 for No. 1A DG and ≤ 0.83 for Nos. 1B, 2A, and 2B DGs. These power factors are representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, Note 2 allows the surveillance to be conducted at a power factor other than ≤ 0.84 for No. 1A DG and ≤ 0.83 for Nos. 1B, 2A, and 2B DGs. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ 0.84 for No. 1A DG and ≤ 0.83 for Nos. 1B, 2A, and 2B DGs results in voltages on the emergency busses that are too high. Conditions can also occur that could result in emergency bus voltages which are too low. Under these conditions, the power factor shall be maintained as close a practicable to 0.84 for No. 1A DG and 0.83 for Nos. 1B, 2A, and 2B DGs while maintaining acceptable voltages on the emergency busses.

Insert 3

~~The 24 month Frequency is adequate to ensure DG OPERABILITY, takes into consideration unit conditions required to perform the surveillance, and is intended to be consistent with expected fuel cycle lengths.~~

SR 3.8.1.12

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This SR demonstrates the DG load response characteristics. This SR is accomplished by tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power.

Consistent with References 10, 3, and 4, 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.

Insert 3

~~The 24 month Frequency is consistent with the Reference 2, Chapter 8.~~

SR 3.8.1.13

This SR demonstrates that DG non-critical protective functions are bypassed on a required actuation signal. This SR is accomplished by verifying the bypass contact changes to the correct state which prevents actuation of the non-critical function. The non-critical protective functions are consistent with References 3 and 4, and Institute of Electrical and Electronic Engineers (IEEE)-387 and are listed in Reference 2, Chapter 8. Verifying the non-critical trips are bypassed will ensure DG operation during a required actuation. The non-critical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. A failure of the electronic governor results in the diesel generator operating in hydraulic mode. This alarm provides the operator with sufficient time to react appropriately. The DG 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 DG.

Insert 3

~~The 24 month Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the surveillance test, 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 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. This Frequency is consistent with Reference 2, Chapter 8.~~

SR 3.8.1.14

This SR ensures that the manual synchronization and load transfer from the DG to the offsite source can be made and that the DG can be returned to ready-to-load status when offsite power is restored. The DG is considered to be in ready-to-load status when the DG is at rated speed and voltage, the output breaker is open and can receive an auto-close signal on bus undervoltage, and the load sequence timers are reset.

Insert 3

~~The Frequency of 24 months takes into consideration unit conditions required to perform the surveillance test.~~

SR 3.8.1.15

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This SR demonstrates the DG operation during a loss of offsite power actuation test signal in conjunction with an ESF (i.e., safety injection) actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system 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.

It is not necessary to energize loads which are dependent on temperature to load (i.e., heat tracing, switchgear HVAC compressor, computer room HVAC compressor). Also, it is acceptable to transfer the instrument AC bus to the non tested train to maintain safe operation of the plant during testing. Loads (both permanent and auto connect) < 15 kW do not require loading onto the diesel since these are insignificant loads for the DG.

Permanently- and auto-connected loads to the emergency diesel generators are defined as follows:

Permanently-Connected Load – Equipment that is not shed by an undervoltage or safety injection actuation signal and is normally operating, i.e., loads that are manually started, selected, or process signal controlled are not considered permanently-connected loads.

Auto-Connected Loads – Emergency equipment required for mitigating the events described in UFSAR Chapter 14 that are energized by loss-of-coolant incident sequencer actions after step zero and within the first minute of emergency diesel generator operation after the initiation of an undervoltage signal.

Insert 3

~~The Frequency of 24 months takes into consideration unit conditions required to perform the surveillance test and is intended to be consistent with an expected fuel cycle length of 24 months.~~

This SR is modified by a Note. The reason for the Note is to minimize mechanical wear and stress on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs.

SR 3.8.1.16

This SR lists the SRs that are applicable to the LCO 3.8.1.c (SRs 3.8.1.1, 3.8.1.2, 3.8.1.3, 3.8.1.5, 3.8.1.6, and 3.8.1.7). Performance of any SR for the LCO 3.8.1.c will satisfy both Unit 1 and Unit 2 requirements for those SRs. Surveillance Requirements 3.8.1.4, 3.8.1.8, 3.8.1.9, 3.8.1.10, 3.8.1.11, 3.8.1.12, 3.8.1.13, 3.8.1.14, and 3.8.1.15, are not required to be performed for the LCO 3.8.1.c. Surveillance Requirement 3.8.1.10 is not required because this SR verifies manual transfer of AC power sources from the normal offsite circuit to the alternate offsite circuit, but only one qualified offsite circuit is necessary for the LCO 3.8.1.c. Surveillance Requirements 3.8.1.4, 3.8.1.11, and 3.1.8.12 are not

F.1

With a Required Action and associated Completion Time not met, or one or more DGs with diesel fuel oil not within limits for reasons other than addressed by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable. "Associated DG(s)" are identified in the Applicability Bases.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the DG FOSTs to support one unit on accident loads and one unit on shutdown loads for seven days. The seven day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

Insert 3

~~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.~~

SR 3.8.3.2

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade (i.e., 2D and 2D low sulfur) and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. Note that further references to American Society for Testing Materials (ASTM) 2D fuel oil include both 2D and 2D low sulfur. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with Reference 3, ASTM D4057-1995;

BASES

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.3

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 from the fuel storage tanks ~~once every 92 days~~ eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, and contaminated fuel oil, and from 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 SR Frequencies are established by Reference 3. This SR is for preventative maintenance.~~ Insert 3 The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed during performance of the surveillance test.

REFERENCES

1. UFSAR
2. ASTM Standards
3. Regulatory Guide 1.137, "Fuel-Oil Systems for Standby Diesel Generators," January 1978

assumed in the battery sizing calculations. ~~The 7 day Frequency is conservative when compared with manufacturer recommendations and Reference 6.~~

Insert 3

SR 3.8.4.2

Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each cell to cell and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The limits established for this SR must be no more than 20% above the resistance as measured during installation or not above the ceiling value established by the manufacturer.

~~The SR Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.~~

SR 3.8.4.3

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

~~The 18 month Frequency is based on engineering judgment. 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 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of cell to cell and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anti-corrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.4.

The connection resistance limits for SR 3.8.4.5 shall be no more than 20% above the resistance as measured during installation, or not above the ceiling value established by the manufacturer.

~~The 18 month Frequency for these SRs is based on engineering judgment. Operating experience has shown that these components usually pass the SRs when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint~~.

Insert 3

SR 3.8.4.6

This SR requires that each battery charger be capable of supplying 400 amps and 125 V for ≥ 30 minutes. These requirements are based on the output rating of the chargers (Reference 1, Chapter 8). According to Reference 7, the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied. The test is performed while supplying normal DC loads or an equivalent or greater dummy load.

~~The SR Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 24 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths~~.

SR 3.8.4.7

A battery service test is a special test of battery capability, as found and with the associated battery charger disconnected, to satisfy the design requirements (battery duty cycle) of the DC source. The test duration must be ≥ 2 hours and battery terminal voltage must be maintained ≥ 105 volts during the test. The discharge rate and test length should correspond to the design accident load (duty) cycle requirements as specified in Reference 1, Chapter 8. A dummy load simulating the emergency loads of the design duty cycle may be used in lieu of the actual emergency loads.

Insert 3

~~The SR Frequency of 24 months is consistent with expected fuel cycle lengths.~~

This SR is modified by a Note. The Note allows the performance of a modified performance discharge test in lieu of a service test. This substitution is acceptable because a modified performance discharge test represents a more severe test of battery capacity than SR 3.8.4.7.

SR 3.8.4.8

A battery performance discharge test is a test of constant current capacity of a battery after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

A battery modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance discharge test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery performance discharge test for the duration of time equal to that of the performance discharge test.

A modified performance discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

The acceptance criteria for this SR are consistent with References 6 and 4. These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

Insert 3

~~The SR Frequency for this test is normally 60 months.~~ If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the SR Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the SR Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's rating. Degradation is indicated, according to Reference 6, when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in Reference 6.

BASES

Continued operation prior to declaring the affected batteries inoperable is permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC channel must be declared inoperable. Additionally, other potentially extreme conditions, such as any Required Action of Condition A and associated Completion Time not met, or average electrolyte temperature of representative cells < 69°F, are also cause for immediately declaring the associated DC channel inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with Reference 2, which recommends regular battery inspections ~~(at least one per month)~~ including voltage, specific gravity, and electrolyte temperature of pilot cells.

Insert 3

SR 3.8.6.2

The ~~quarterly~~ inspection of specific gravity and voltage is consistent with Reference 2.

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells is > 69°F is consistent with a recommendation of Reference 2, which states that the temperature of electrolytes in representative cells should be determined ~~on a quarterly basis~~. The temperature is also high enough to supply the required capacity.

from its 120 VAC bus powered by an ESF motor control center through a regulating transformer.

Required Action A.1 is modified by a Note, which states to enter the applicable conditions and Required Actions of LCO 3.8.9, when Condition A is entered with one AC vital bus de-energized. This ensures the vital bus is re-energized within two hours.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its constant voltage source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within six hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This SR verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. ~~The seven day Frequency~~

Insert 3

BASES

~~takes into account the redundant capability of the inverters and other indications available in the Control Room that alert the operator to inverter malfunctions~~.

REFERENCES 1. UFSAR

BASES

voltage output ensures that the required power is readily available for the instrumentation connected to the AC vital buses. ~~The seven day Frequency takes into account the redundant capability of the inverters and other indications available in the Control Room that alert the operator to inverter malfunctions.~~

Insert 3

REFERENCES 1. UFSAR

BASES

Insert 3

loads connected to these buses. ~~The seven day Frequency takes into account the redundant capability of the AC, DC, and AC vital bus electrical power distribution subsystems, and other indications available in the Control Room that alert the operator to subsystem malfunctions~~.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.93, "Availability of Electric Power Sources," December 1974

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required shutdown cooling (SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.3 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the SDC ACTIONS would not be entered. Therefore, Required Action A.2.4 is provided to direct declaring SDC inoperable, which results in taking the appropriate SDC actions. The SDC subsystem(s) declared inoperable and not in operation as a result of not meeting this LCO, may be used if needed. However, the appropriate actions are still required to be taken.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This SR verifies that the AC, DC, and AC vital bus Electrical Power Distribution System is functioning properly, with all the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. ~~The seven day Frequency takes into account the redundant capability of the electrical power distribution subsystems, and other indications available in the Control Room that alert the operator to subsystem malfunctions~~.

Insert 3

BASES

depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This Surveillance Requirement (SR) ensures the coolant boron concentration in the RCS and the refueling pool is within the COLR limits. The coolant boron concentration in each volume is determined periodically by chemical analysis.

Insert 3

~~A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate~~.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR)

BASES

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this period.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

Surveillance Requirement 3.9.2.1 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 3.3.1.~~

Insert 3

SR 3.9.2.2

Surveillance Requirement 3.9.2.2 is the performance of a CHANNEL CALIBRATION ~~every 24 months~~. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. This is because generating a meaningful test signal is difficult; the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. ~~This Frequency is the same as that employed for the same channels in the other applicable MODEs.~~

REFERENCES

1. UFSAR

containment atmosphere to the outside atmosphere through a filtered or unfiltered pathway not in the required status, (including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open) the unit must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending movement of irradiated fuel assemblies within the Containment Structure. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This SR demonstrates that each of the containment penetrations required to be in its closed position, is in that position. The surveillance test on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also, the surveillance test will demonstrate that each purge and exhaust valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE automatic Containment Purge Valve Isolation System.

~~The surveillance test is performed every seven days during movement of irradiated fuel assemblies within the Containment Structure. The surveillance test interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance test before the start of refueling operations will provide two or three verifications during the applicable period for this LCO. As such, this SR ensures that a postulated fuel handling accident, that releases fission product radioactivity within the Containment Structure, will not result in a release of fission product radioactivity to the environment in excess of those described in Reference 1.~~

Insert 3

SR 3.9.3.2

This SR demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. ~~The once each refueling outage Frequency, maintains consistency with other similar Engineered Safety Features Actuation System instrumentation and valve testing.~~

BASES

~~requirements. However, in order to ensure the SR Frequency is satisfied, this surveillance test is typically performed once per refueling outage prior to the start of movement of irradiated fuel assemblies within Containment. In LCO 3.3.7, the Containment Radiation Signal System requires a CHANNEL CHECK every 12 hours and a CHANNEL FUNCTIONAL TEST every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 24 months during refueling on a STAGGERED TEST BASIS. Surveillance Requirement 3.6.3.4 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements.~~ These surveillance tests performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the Containment Structure.

REFERENCES

1. UFSAR, Section 14.18, "Fuel Handling Incident"

BASES

With SDC loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensure that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of four hours allows fixing of most SDC problems and is reasonable, based on the low probability of the coolant boiling in that time.

The emergency air lock temporary closure device cannot be credited for containment closure for a loss of shutdown cooling event. At least one door in the emergency air lock must be closed to satisfy this action statement.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This SR demonstrates that the SDC loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability, and to prevent thermal and boron stratification in the core. ~~The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the Control Room for monitoring the SDC System~~.

Insert 3

REFERENCES

1. UFSAR, Section 9.2, "Shutdown Cooling System"

The Completion Time of four hours allows fixing of most SDC problems and is reasonable, based on the low probability of the coolant boiling in that time.

The emergency air lock temporary closure device cannot be credited for containment closure for a loss of shutdown cooling event. At least one door in the emergency air lock must be closed to satisfy this action statement.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This SR demonstrates that one SDC loop is operating and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. This SR also demonstrates that the other SDC loop is OPERABLE.

In addition, during operation of the SDC loop with the water level in the vicinity of the reactor vessel nozzles, the SDC loop flow rate determination must also consider the SDC pump suction requirements. ~~The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the SDC System in the Control Room.~~

~~Verification that the required loops are OPERABLE and in operation ensures that loops can be placed in operation as needed, to maintain decay heat and retain forced circulation. The Frequency of 12 hours is considered reasonable, since other administrative controls are available and have proven to be acceptable by operating experience.~~

Insert 3

SR 3.9.5.2

This SR demonstrates that the SDC loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. ~~The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available for the operator in the Control Room for monitoring the SDC System.~~

BASES

SR 3.9.5.3

Verification that the required pump and valves are OPERABLE ensures that an additional SDC loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump and valves. Insert 3 ~~The Frequency of seven days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

REFERENCES

1. UFSAR, Section 9.2, "Shutdown Cooling System"

BASES

	Structure are within the acceptable limits given in Reference 2.
APPLICABILITY	LCO 3.9.6 is applicable when moving irradiated fuel assemblies in the Containment Structure. The LCO minimizes the possibility of a fuel handling accident in the Containment Structure that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in the Containment Structure, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.13.
ACTIONS	A.1 With a water level of < 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel, all operations involving movement of irradiated fuel assemblies, shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement shall not stop the movement of a component to a safe position.
SURVEILLANCE REQUIREMENTS	SR 3.9.6.1 Verification of a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated in the reactor vessel ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the irradiated fuel assemblies seated in the reactor vessel limits the consequences of damaged fuel rods, that are postulated to result from a fuel handling accident inside the Containment Structure (Reference 2). Insert 3 ~~The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water, and the normal procedural controls of valve positions which make significant unplanned level changes unlikely.~~

ATTACHMENT (5)

COMPARISON MATRIX

Technical Specification Title/Surveillance Description	TSTF-425	CCNPP
Shutdown Margin (SDM)	**3.1.1**	**3.1.1**
Verify SDM within limits	SR 3.1.1.1	SR 3.1.1.1
Reactivity Balance	**3.1.2**	-----
CEA Alignment	**3.1.4**	**3.1.4**
Verify indicated position within 7 inches	SR 3.1.4.1	SR 3.1.4.1
Verify motion inhibit is Operable	SR 3.1.4.2	SR 3.1.4.2
Verify deviation circuit is Operable	SR 3.1.4.3	SR 3.1.4.3
Verify CEA freedom of movement	SR 3.1.4.4	SR 3.1.4.4
Perform Channel Functional Test	SR 3.1.4.5	SR 3.1.4.5
Shutdown CEA Insertion Limits	**3.1.5**	**3.1.5**
Verify CEA is withdrawn	SR 3.1.5.1	SR 3.1.5.1
Regulating CEA Insertion Limits	**3.1.6**	**3.1.6**
Verify CEA group position is within limits	SR 3.1.6.1	SR 3.1.6.1
Verify CEA insertion times	SR 3.1.6.2	SR 3.1.6.2
Verify PDIL alarm circuit is Operable	SR 3.1.6.3	SR 3.1.6.3
STE-SDM	**3.1.7**	**3.1.7**
Verify CEA insertion is within acceptance criteria	SR 3.1.7.1	SR 3.1.7.1
STE-Modes 1 and 2	**3.1.8**	**3.1.8**
Verify Thermal Power is within test power plateau	SR 3.1.8.l	SR 3.1.8.1
LHR	**3.2.1**	**3.2.1**
Verify ASI alarm setpoints	SR 3.2.1.1	SR 3.2.1.2
Verify incore detector local power density alarms	SR 3.2.1.2	SR 3.2.1.3
Verify incore local power density alarm setpoints	SR 3.2.1 3	SR 3.2.1.4
Fxy	**3.2.2**	-----
Fr	**3.2.3**	**3.2.3**
Verify value of Fr	SR 3.2.3.1	SR 3.2.3.1
Tq	**3.2.4**	**3.2.4**
Verify value of Tq	SR 3.2.4.1	SR 3.2.4.1
ASI	**3.2.5**	**3.2.5**
Verify ASI is within limits	SR 3.2.5.1	SR 3.2.5.1
RPS Instrumentation –Operating	**3.3.1**	**3.3.1**
Perform Channel Check	SR 3.3.1.1	SR 3.3.1.1
Perform calibration of excore and dT power channels	SR 3.3.1.2	SR 3.3.1.2
Calibrate power range excores using incore detectors	SR 3.3.1.3	SR 3.3.1.3
Perform Channel Functional Test	SR 3.3.1.4	SR 3.3.1.4
Perform Channel Calibration on excore power range channels	SR 3.3.1.5	SR 3.3.1.5
Perform Channel Functional Test on automatic bypass removal	-----	SR 3.3.1.7
Perform Channel Calibration of each RPS channel	SR 3.3.1.8	SR 3.3.1.8
Verify RPS response times	SR 3.3.1.9	SR 3.3.1.9
RPS Instrumentation – Shutdown	**3.3.2**	**3.3.2**
Perform Channel Check of wide range power channel	SR 3.3.2.1	SR 3.3.2.1
Perform Channel Functional Test of power rate of change trip	SR 3.3.2.2	-----
Perform Channel Functional Test of automatic bypass removal	SR 3.3.2.3	SR 3.3.2.3
Perform Channel Calibration, including bypass functions	SR 3.3.2.4	SR 3.3.2.4
RPS Logic and Trip Initiation	**3.3.3**	**3.3.3**
Perform Channel Functional Test on RTCB channel	SR 3.3.3.1	SR 3.3.3.1

Technical Specification Title/Surveillance Description	TSTF-425	CCNPP
Perform Channel Functional Test on RPS logic	SR 3.3.3.2	SR 3.3.3.2
Perform Channel Functional Test with undervoltage and shunt trips	SR 3.3.3.4	-----
ESFAS Instrumentation	**3.3.4**	**3.3.4**
Perform Channel Check on each ESFAS channel	SR 3.3.4.1	SR 3.3.4.1
Perform Channel Functional Test on each ESFAS channel	SR 3.3.4.2	SR 3.3.4.2
Perform Channel Functional Test on automatic block removal	-----	SR 3.3.4.3
Perform Channel Calibration of ESFAS channels, including block removal	SR 3.3.4.4	SR 3.3.4.4
Verify ESF response time in limits	SR 3.3.4.5	SR 3.3.4.5
ESFAS Logic and Manual Trip/Actuation	**3.3.5**	**3.3.5**
Perform Channel Functional Test on ESFAS Logic channel	SR 3.3.5.1	SR 3.3.5.1
Perform Channel Functional Test on ESFAS trip/actuation channel	SR 3.3.5.2	SR 3.3.5.2
DG-LOVS	**3.3.6**	**3.3.6**
Perform Channel Check	SR 3.3.6.1	-----
Perform Channel Functional Test	SR 3.3.6.2	SR 3.3.6.1
Perform Channel Calibration	SR 3.3.6.3	SR 3.3.6.2
CPIS/CRS	**3.3.7**	**3.3.7**
Perform Channel Check on radiation monitors	SR 3.3.7.1	SR 3.3.7.1
Perform Channel Functional Test on radiation monitor channel	SR 3.3.7.2	SR 3.3.7.3
Perform Channel Functional Test on actuation logic channel	SR 3.3.7.3	SR 3.3.7.2
Perform Channel Calibration on radiation monitor channel	SR 3.3.7.4	SR 3.3.7.4
Perform Channel Functional Test on manual trip/actuation channel	SR 3.3.7.5	SR 3.3.7.5
Verify response time is within limits	SR 3.3.7.6	SR 3.3.7.6
CRIS/CRRS	**3.3.8**	**3.3.8**
Perform Channel Check on radiation monitor channel	SR 3.3.8.1	SR 3.3.8.1
Perform Channel Functional Test on the radiation monitor channel	SR 3.3.8.2	SR 3.3.8.2
Perform Channel Functional Test on actuation logic channel	SR 3.3.8.3	-----
Perform Channel Calibration on the radiation monitor channel	SR 3.3.8.4	SR 3.3.8.3
Perform Channel Functional Test on the manual trip channel	SR 3.3.8.5	-----
Verify response times are within limits	SR 3.3.8.6	-----
CVCS Isolation Signal	**3.3.9**	**3.3.9**
Perform Channel Check	SR 3.3.9.1	SR 3.3.9.1
Perform Channel Functional Test on CVCS isolation channels	SR 3.3.9.2	SR 3.3.9.2
Perform Channel Calibration on CVCS sensor channels	SR 3.3.9.3	SR 3.3.9.3
Verify response time is within limits	-----	SR 3.3.9.4
Shield Building Filtration Actuation Signal	**3.3.10**	**-----**
PAM Instrumentation	**3.3.11**	**3.3.10**
Perform Channel Check for normally energized channels	SR 3.3.11.1	SR 3.3.10.1
Perform Channel Calibration	SR 3.3.11.2	SR 3.3.10.3
Remote Shutdown System/Instrumentation	**3.3.12**	**3.3.11**
Perform Channel Check for normally energized channels	SR 3.3.12.1	SR 3.3.11.1
Verify control circuit and transfer switch works	SR 3.3.12.2	-----
Perform Channel Calibration for each channel	SR 3.3.12.3	SR 3.3.11.2

Technical Specification Title/Surveillance Description	TSTF-425	CCNPP
Perform Channel Functional Test	SR 3.3.12.4	-----
Power Monitoring Channels/Wide Range Logarithmic Neutron Flux Monitor Channels	**3.3.13**	**3.3.12**
Perform Channel Check	SR 3.3.13.1	SR 3.3.12.1
Perform Channel Functional Test	SR 3.3.13.2	-----
Perform Channel Calibration	SR 3.3.13.3	SR 3.3.12.3
RCS Pressure, Temperature and Flow DNBR Limits	**3.4.1**	**3.4.1**
Verify pressurizer pressure is within limits	SR 3.4.1.1	SR 3.4.1.1
Verify cold leg temperatures are within limits	SR 3.4.1.2	SR 3.4.1.2
Verify RCS total flow	SR 3.4.1.3	SR 3.4.1.3
Verify heat balance/measured RCS flow is within limits	SR 3.4.1.4	SR 3.4.1.4
RCS Minimum Temperature for Criticality	**3.4.2**	-----
RCS P/T Limits	**3.4.3**	**3.4.3**
Verify RCS temperature, pressure, rates are within limits	SR 3.4.3.1	SR 3.4.3.1
RCS Loops - Modes 1 and 2	**3.4.4**	**3.4.4**
Verify RCS loops in operation	SR 3.4.4.1	SR 3.4.4.1
RCS Loops - Mode 3	**3.4.5**	**3.4.5**
Verify RCS loop is in operation	SR 3.4.5.1	SR 3.4.5.1
Verify secondary side SG water level	SR 3.4.5.2	SR 3.4.5.2
Verify correct breaker alignment	SR 3.4.5.3	SR 3.4.5.3
RCS Loops - Mode 4	**3.4.6**	**3.4.6**
Verify one RCS/SDC in operation	SR 3.4.6.1	SR 3.4.6.1
Verify secondary side SG water level	SR 3.4.6.2	SR 3.4.6.2
Verify correct breaker alignment	SR 3.4.6.3	SR 3.4.6.3
RCS Loops - Mode 5, Loops Filled	**3.4.7**	**3.4.7**
Verify SDC train is in operation	SR 3.4.7.1	SR 3.4.7.1
Verify secondary side SG water level	SR 3.4.7.2	SR 3.4.7.2
Verify correct breaker alignment	SR 3.4.7.3	SR 3.4.7.3
RCS Loops - Mode 5, Loops not Filled	**3.4.8**	**3.4.8**
Verify SDC train is in operation	SR 3.4.8.1	SR 3.4.8.1
Verify correct breaker alignment	SR 3.4.8.2	SR 3.4.8.2
Pressurizer	**3.4.9**	**3.4.9**
Verify pressurizer water level	SR 3.4.9.1	SR 3.4.9.1
Verify capacity of heaters	SR 3.4.9.2	SR 3.4.9.2
Verify heaters are powered from emergency power	SR 3.4.9.3	-----
Pressurizer PORVs	**3.4.11**	**3.4.11**
Perform Channel Functional Test of PORVs	-----	SR 3.4.11.1
Perform cycle of block valves	SR 3.4.11.1	SR 3.4.11.2
Perform cycle of PORVs	SR 3.4.11.2	SR 3.4.11.3
Perform Channel Calibration of PORVs	-----	SR 3.4.11.4
Perform cycle of control valve and check valve on PORV	SR 3.4.11.3	-----

Technical Specification Title/Surveillance Description	TSTF-425	CCNPP
Verify PORVs and block valves are powered from emergency power	SR 3.4.11.4	-----
LTOP System	**3.4.12**	**3.4.12**
Verify one HPSI capable of RCS injection	SR 3.4.12.1	SR 3.4.12.1
Verify one charging pump capable of RCS injection	SR 3.4.12.2	-----
Verify SITs are isolated	SR 3.4.12.3	-----
Verify HPSI MOVs are capable of manual alignment	-----	SR 3.4.12.2
Verify RCS vent is open	SR 3.4.12.4	SR 3.4.12.3
Verify PORV block valve is open	SR 3.4.12.5	SR 3.4.12.4
Perform Channel Functional Test for PORVs	SR 3.4.12.6	SR 3.4.12.5
Perform Channel Calibration on PORVs	SR 3.4.12.7	SR 3.4.12.6
RCS Operational Leakage	**3.4.13**	**3.4.13**
Verify leakage is within limits	SR 3.4.13.1	SR 3.4.13.1
Verify primary to secondary leakage is within limits	SR 3.4.13.2	SR 3.4.13.2
RCS PIV Leakage	**3.4.14**	-----
RCS Leakage Detection Instrumentation	**3.4.15**	**3.4.14**
Perform Channel Check of radiation monitor	SR 3.4.15.1	SR 3.4.14.1
Perform Channel Functional Test of radiation monitor	SR 3.4.15.2	SR 3.4.14.2
Perform Channel Calibration of sump monitor	SR 3.4.15.3	SR 3.4.14.3
Perform Channel Calibration of radiation monitor	SR 3.4.15.4	SR 3.4.14.4
Perform Channel Calibration of condensate flow monitor	SR 3.4.15.5	-----
RCS Specific Activity	**3.4.16**	**3.4.15**
Verify gross reactor coolant activity	SR 3.4.16.1	SR 3.4.15.1
Verify reactor coolant specific activity	SR 3.4.16.2	SR 3.4.15.2
Determine E	SR 3.4.16.3	SR 3.4.15.3
STE-RCS Loops, Mode 2	**3.4.17**	**3.4.16**
Verify thermal power	SR 3.4.17.1	SR 3.4.16.1
STE-RCS Loops, Modes 4 & 5	-----	**3.4.17**
Verify charging pumps are de-energized	-----	SR 3.4.17.2
Verify charging flow path is isolated	-----	SR 3.4.17.3

Technical Specification Title/Surveillance Description	TSTF-425	CCNPP
Perform SR 3.1.1.1	-----	SR 3.4.17.4
SITs	**3.5.1**	**3.5.1**
Verify isolation valves are open	SR 3.5.1.1	SR 3.5.1.1
Verify borated water volume in SITs	SR 3.5.1.2	SR 3.5.1.2
Verify nitrogen pressure in SITs	SR 3.5.1.3	SR 3.5.1.3
Verify boron concentration in SITs	SR 3.5.1.4	SR 3.5.1.4
Verify power is removed from SIT isolation valves	SR 3.5.1.5	SR 3.5.1.5
ECCS-Operating	**3.5.2**	**3.5.2**
Verify specified valve position	SR 3.5.2.1	SR 3.5.2.1
Verify valves are in the correct position	SR 3.5.2.2	SR 3.5.2.2
Verify ECCS piping is full of water	SR 3.5.2.3	-----
Verify valves actuate to correct position	SR 3.5.2.6	SR 3.5.2.5
Verify ECCS pumps start on signal	SR 3.5.2.7	SR 3.5.2.6
Verify LPSI pump stops on signal	SR 3.5.2.8	SR 3.5.2.7
Verify throttle valves stop in correct position	SR 3.5.2.9	-----
Verify containment sump is not blocked by debris	SR 3.5.2.10	SR 3.5.2.8
Verify SDC interlock works	-----	SR 3.5.2.9
RWT	**3.5.4**	**3.5.4**
Verify water temperature	SR 3.5.4.1	SR 3.5.4.1
Verify water temperature	SR 3.5.4.1	SR 3.5.4.2
Verify water volume	SR 3.5.4.2	SR 3.5.4.3
Verify boron concentration	SR 3.5.4.3	SR 3.5.4.4
TSP/STB	**3.5.5**	**3.5.5**
Verify baskets contain TSP/STB	SR 3.5.5.1	SR 3.5.5.1
Verify pH adjustment of water	SR 3.5.5.2	SR 3.5.5.2
Containment Air Locks	**3.6.2**	**3.6.2**
Verify only one door open at a time	SR 3.6.2.2	SR 3.6.2.2
Containment Isolation Valves	**3.6.3**	**3.6.3**
Verify purge valves are closed	SR 3.6.3.1	-----
Verify vent valves are closed	SR 3.6.3.2	SR 3.6.3.1
Verify isolation valves are closed	SR 3.6.3.3	SR 3.6.3.2
Verify isolation time of valves	SR 3.6.3.5	-----
Perform leak rate test of purge valves	SR 3.6.3.6	-----
Verify valves actuate to correct position	SR 3.6.3.7	SR 3.6.3.5
Verify purge valve is blocked	SR 3.6.3.8	-----
Containment Pressure	**3.6.4**	**3.6.4**
Verify pressure is within limits	SR 3.6.4.1	SR 3.6.4.1
Containment Air Temperature	**3.6.5**	**3.6.5**
Verify temperature is within limits	SR 3.6.5.1	SR 3.6.5.1
Containment Spray and Cooling System	**3.6.6A**	**3.6.6**
Verify containment spray valves are in the correct position	SR 3.6.6A.1	SR 3.6.6.1
Operate cooling train fan	SR 3.6.6A.2	SR 3.6.6.2
Verify cooling water flow rate	SR 3.6.6A.3	SR 3.6.6.3
Verify containment spray pipe is full of water	SR 3.6.6A.4	-----
Verify valves actuate to their correct position	SR 3.6.6A.6	SR 3.6.6.5

Technical Specification Title/Surveillance Description	TSTF-425	CCNPP
Verify containment spray pump starts	SR 3.6.6A.7	SR 3.6.6.6
Verify cooling train starts	SR 3.6.6A.8	SR 3.6.6.7
Verify spray nozzle is unobstructed	SR 3.6.6A.9	-----
Spray Additive System	**3.6.7**	-----
Shield Building Exhaust Air Cleanup System	**3.6.8**	-----
HMS	**3.6.9**	-----
Iodine Cleanup System/Iodine Removal System	**3.6.10**	**3.6.8**
Operate each train	SR 3.6.10.1	SR 3.6.8.1
Verify each train actuates	SR 3.6.10.3	SR 3.6.8.3
Verify bypass damper can be opened	SR 3.6.10.4	----
Shield Building	**3.6.11**	-----
MSIVs	**3.7.2**	**3.7.2**
MFIVs	**3.7.3**	**3.7.15**
ADVs	**3.7.4**	-----
AFW System	**3.7.5**	**3.7.3**
Verify valves are in the correct position	SR 3.7.5.1	SR 3.7.3.1
Verify valves actuate to the correct position	SR 3.7.5.3	SR 3.7.3.4
Verify the AFW pumps start automatically	SR 3.7.5.4	SR 3.7.3.5
Verify the AFW system can provide minimum flow	-----	SR 3.7.3.6
Condensate Storage Tank	**3.7.6**	**3.7.4**
Verify CST level/volume	SR 3.7.6.1	SR 3.7.4.1
CCW/CC System	**3.7.7**	**3.7.5**
Verify valves are in the correct position	SR 3.7.7.1	SR 3.7.5.1
Verify valves actuate to the correct position	SR 3.7.7.2	SR 3.7.5.2
Verify pumps start automatically	SR 3.7.7.3	SR 3.7.5.3
SWS/SRW	**3.7.8**	**3.7.6**
Verify valves are in the correct position	SR 3.7.8.1	SR 3.7.6.1
Verify valves actuate to the correct position	SR 3.7.8.2	SR 3.7.6.2
Verify pumps start automatically	SR 3.7.8.3	SR 3.7.6.3
SWS/SW	**3.7.8**	**3.7.7**
Verify valves are in the correct position	SR 3.7.8.1	SR 3.7.7.1
Verify valves actuate to the correct position	SR 3.7.8.2	SR 3.7.7.2
Verify pumps start automatically	SR 3.7.8.3	SR 3.7.7.3
UHS	**3.7.9**	-----
ECW	**3.7.10**	-----
CREACS/CREVS	**3.7.11**	**3.7.8**
Operate filter train	SR 3.7.11.1	SR 3.7.8.1
Verify each train actuates on a signal	SR 3.7.11.3	SR 3.7.8.3
Verify positive pressure can be maintained	SR 3.7.11.4	-----
CREATS/CRETS	**3.7.12**	**3.7.9**
Verify heat load can be removed/temperature can be maintained	SR 3.7.12.1	SR 3.7.9.1
ECCS PREACS	**3.7.13**	-----
FBACS/SFPEVS	**3.7.14**	**3.7.11**
Operate each train	SR 3.7.14.1	-----
Verify a train is in operation	-----	SR 3.7.11.1

Technical Specification Title/Surveillance Description	TSTF-425	CCNPP
Verify each train actuates on a signal	SR 3.7.14.3	-----
Verify a train can maintain a negative pressure	SR 3.7.14.4	SR 3.7.11.3
Verify the bypass damper can be opened	SR 3.7.14.5	-----
PREACS/PREVS	**3.7.15**	**3.7.12**
Operate each train	SR 3.7.15.1	SR 3.7.12.1
Verify each train actuates on a signal	SR 3.7.15.3	SR 3.7.12.3
Verify a train can maintain a negative pressure	SR 3.7.15.4	-----
Verify the bypass damper can be opened	SR 3.7.15.5	-----
Fuel Storage Pool Water Level/ SFP Water Level	**3.7.16**	**3.7.13**
Verify water level in pool	SR 3.7.16.1	SR 3.7.13.1
FSP Boron Concentration/SFP Boron Concentration	**3.7.17**	**3.7.16**
Verify boron concentration in pool	SR 3.7.17.1	SR 3.7.16.1
Secondary Specific Activity	**3.7.19**	**3.7.14**
Verify secondary specific activity is within limits	SR 3.7.19.1	SR 3.7.14.1
AC Sources-Operating	**3.8.1**	**3.8.1**
Verify correct breaker alignment	SR 3.8.1.1	SR 3.8.1.1
Verify correct breaker alignment	SR 3.8.1.1	SR 3.8.1.2
Verify DG standby start	SR 3.8.1.2	SR 3.8.1.3
Verify DG is loaded and operate for 1 hour	SR 3.8.1.3	SR 3.8.1.4
Verify day tank volume	SR 3.8.1.4	SR 3.8.1.5
Check for water in the day tank	SR 3.8.1.5	SR 3.8.1.6
Verify fuel oil transfer system operates	SR 3.8.1.6	SR 3.8.1.7
Verify DG standby start (fast start)	SR 3.8.1.7	SR 3.8.1.9
Verify manual transfer of AC sources	SR 3.8.1.8	SR 3.8.1.10
Verify single load reject	SR 3.8.1.9	SR 3.8.1.12
Verify full load reject	SR 3.8.1.10	-----
Verify DG start and operation on a loss of offsite power	SR 3.8.1.11	-----
Verify DG start and operation on an ESF signal	SR 3.8.1.12	-----
Verify non-critical trips are bypassed	SR 3.8.1.13	SR 3.8.1.13
Verify DG operation for 24/1 hour	SR 3.8.1.14	SR 3.8.1.11
Verify hot restart of the DG	SR 3.8.1.15	-----
Verify DG synchronizes to offsite power	SR 3.8.1.16	SR 3.8.1.14
Verify ESF signal overrides DG test mode	SR 3.8.1.17	-----
Verify load sequencer operation	SR 3.8.1.18	SR 3.8.1.8
Verify DG start and operation on an ESF/LOOP signal	SR 3.8.1.19	SR 3.8.1.15

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Verify DG operation during simultaneous start	SR 3.8.1.20	-----
Diesel Fuel Oil, Lube Oil and Starting Air/Diesel Fuel Oil	**3.8.3**	**3.8.3**
Verify volume of diesel fuel oil	SR 3.8.3.1	SR 3.8.3.1
Verify lube oil inventory	SR 3.8.3.2	-----
Verify air start receiver pressure	SR 3.8.3.4	-----
Remove accumulated water from storage tank	SR 3.8.3.5	SR 3.8.3.3
DC Sources – Operating	**3.8.4**	**3.8.4**
Verify battery terminal voltage	SR 3.8.4.1	SR 3.8.4.1
Verify battery charger provides adequate voltage	SR 3.8.4.2	SR 3.8.4.6
Verify battery capacity	SR 3.8.4.3	SR 3.8.4.7
Verify no visible corrosion	-----	SR 3.8.4.2
Verify no physical damage	-----	SR 3.8.4.3
Remove visible corrosion	-----	SR 3.8.4.4
Verify battery connection resistance	-----	SR 3.8.4.5
Battery Parameters/Battery Cell Parameters	**3.8.6**	**3.8.6**
Verify float current/voltage	SR 3.8.6.1	SR 3.8.6.1
Verify battery cell voltage	SR 3.8.6.2	SR 3.8.6.2
Verify electrolyte level	SR 3.8.6.3	SR 3.8.6.1
Verify battery temperature	SR 3.8.6.4	SR 3.8.6.3
Verify cell voltage	SR 3.8.6.5	SR 3.8.6.2
Verify battery capacity – discharge test	SR 3.8.6.6	SR 3.8.4.8
Inverters-Operating	**3.8.7**	**3.8.7**
Verify correct voltage	SR 3.8.7.1	SR 3.8.7.1
Inverters-Shutdown	**3.8.8**	**3.8.8**
Verify correct voltage	SR 3.8.8.1	SR 3.8.8.1
Distribution Systems-Operating	**3.8.9**	**3.8.9**
Verify correct breaker alignments	SR 3.8.9.1	SR 3.8.9.1
Distribution Systems-Shutdown	**3.8.10**	**3.8.10**
Verify correct breaker alignments	SR 3.8.10.1	SR 3.8.10.1
Boron Concentration	**3.9.1**	**3.9.1**
Verify boron concentration	SR 3.9.1.1	SR 3.9.1.1
Nuclear Instrumentation	**3.9.2**	**3.9.2**
Perform Channel Check	SR 3.9.2.1	SR 3.9.2.1
Perform Channel Calibration	SR 3.9.2.2	SR 3.9.2.2
Containment Penetrations	**3.9.3**	**3.9.3**
Verify containment penetrations are in correct status	SR 3.9.3.1	SR 3.9.3.1
Verify containment purge actuates on a signal	SR 3.9.3.2	SR 3.9.3.2
SDC and Coolant Circulation-High Water Level	**3.9.4**	**3.9.4**
Verify one SDC loop is in operation	SR 3.9.4.1	SR 3.9.4.1
SDC and Coolant Circulation-Low Water Level	**3.9.5**	**3.9.5**
Verify SDC loops are operable	SR 3.9.5.1	SR 3.9.5.1
Verify correct breaker alignment	SR 3.9.5.2	SR 3.9.5.3
Verify flow rate of SDC loop	-----	SR 3.9.5.2
Refueling Water Level/Refueling Pool Water Level	**3.9.6**	**3.9.6**
Verify water level	SR 3.9.6.1	SR 3.9.6.1