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10 CFR 50.90

1CAN031801

March 12, 2018

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

SUBJECT: Application for Technical Specification Change Regarding Risk-Informed
Justification for the Relocation of Specific Surveillance Frequency
Requirements to a Licensee Controlled Program (TSTF-425)
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

REFERENCES: NUREG-1430, "Standard Technical Specifications Babcock and Wilcox
Plants," Revision 4, April 2012 (ML12100A177)

Dear Sir or Madam:

In accordance with the provisions of Title 10 of the Code of Federal Regulations (10 CFR Part 50.90), "Application for amendment of license, construction permit, or early site permit," Entergy Operations, Inc. (Entergy) is submitting a request for an amendment to the technical specifications (TSs) for Arkansas Nuclear One, Unit 1 (ANO-1).

The proposed amendment would modify the ANO-1 TSs by relocating specific surveillance frequencies to a licensee-controlled program with the implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specification Initiative 5B, Risk-Informed Method for Control of Surveillance Frequencies."

Attachment 1 provides a description of the proposed change, the requested confirmation of applicability, and plant-specific verifications. Attachment 2 provides documentation of the probabilistic risk assessment (PRA) technical adequacy. Attachment 3 provides the existing TS pages marked up to show the proposed change. Attachment 4 provides revised (clean) TS pages. Attachment 5 provides the proposed TS Bases changes for information only. Attachment 6 provides a proposed No Significant Hazards Consideration, consistent with that published in the Federal Register on July 6, 2009 (74 FR 32000). Attachment 7 provides a cross reference table that correlates ANO-1 TS surveillance requirement numbers to the NUREG-1430 (Reference) TS surveillance requirement numbers.

Entergy requests approval of the proposed license amendment by April 1, 2019, with the amendment being implemented within 90 days.

No new regulatory commitments are made in this submittal.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," a copy of this application, with attachments, is being provided to designated Arkansas State Official.

If you should have any questions regarding this submittal, please contact Stephenie Pyle, Manager, Regulatory Assurance, at 479.858.4704.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on March 12, 2018.

Sincerely,

ORIGINAL SIGNED BY RICHARD L. ANDERSON

RLA/dbb

Attachments:

1. Description and Assessment
2. Documentation of PRA Technical Adequacy
3. Proposed Technical Specification Changes (markup)
4. Revised Technical Specification Pages
5. Proposed Technical Specification Bases Changes (Information only)
6. Proposed No Significant Hazards Consideration
7. Arkansas Nuclear One, Unit 1 (ANO-1) TS Surveillance Requirements (SRs) to NUREG-1430 SRs Cross-Reference

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ATTACHMENT 1

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DESCRIPTION AND ASSESSMENT

1.0 Description

The proposed amendment would modify the Arkansas Nuclear One, Unit 1 (ANO-1), Technical Specifications (TSs) by relocating specific surveillance frequencies to a licensee-controlled program with the adoption of Technical Specification Task Force (TSTF)-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b." Additionally, the change would add a new program, the Surveillance Frequency Control Program (SFCP), to TS Section 5.5, "Programs and Manuals."

The changes are consistent with NRC approved Industry/TSTF Standard Technical Specifications (STS) change TSTF-425, Revision 3, (ML090850642). The Federal Register Notice published on July 6, 2009 (74 FR 31996) announced the availability of this TS improvement.

2.0 Assessment

2.1 Applicability of Published Safety Evaluation

Entergy Operations, Inc. (Entergy) has reviewed the safety evaluation provided in Federal Register Notice 74 FR 31996, dated July 6, 2009. This review included the NRC staff's model safety evaluation (SE), TSTF-425, Revision 3, and the requirements specified in Nuclear Energy Institute (NEI) 04-10, Revision 1 (ML071360456).

Attachment 2 includes Entergy's documentation regarding the ANO-1 probabilistic risk assessment (PRA) technical adequacy consistent with the requirements of Regulatory Guide 1.200, Revision 2 (ML090410014), Section 4.2, and describes any PRA models without NRC-endorsed standards, including documentation of the quality characteristics of those models in accordance with Regulatory Guide 1.200.

Entergy has concluded that the justifications presented in the TSTF proposal and the model SE prepared by the NRC staff are applicable to ANO-1 and justify this amendment to incorporate the changes to the ANO-1 TSs.

2.2 Optional Changes and Variations

The proposed amendment is consistent with the STS changes described in TSTF-425, Revision 3; however, Entergy proposes variations or deviations from TSTF-425, as identified below, which include differing TS surveillance numbers.

1. ANO-1 Surveillance Requirements (SRs) with SR numbers that differ from the corresponding TSTF-425 SRs are administrative deviations from TSTF-425 with no impact on the NRC's model SE dated July 6, 2009 (74 FR 32001).
2. The NUREG 1430 TSTF-425 markups add the new Surveillance Frequency Control Program as TS 5.5.18. ANO-1 is adopting this new program as TS 5.5.8, which is currently unused. This is an administrative deviation from TSTF-425 with no impact on the NRC's model SE dated July 6, 2009 (74 FR 32001).

3. For NUREG-1430 SRs not contained in the ANO-1 TS, the corresponding mark-ups included in TSTF-425 for these SRs are not applicable to ANO-1. This is an administrative deviation from TSTF-425 with no impact on the NRC's model SE dated July 6, 2009 (74 FR 32001).
4. Periodic frequencies associated with ANO-1 TSs 5.5.2 and 5.5.13 are included in the scope of this amendment that are not identified for relocation in TSTF-425, Revision 3.

The first sentence of TS 5.5.2.b is revised as follows (deleted text in ~~strikeout~~ and added text in *italics*):

“Integrated leak test requirements for each system at ~~least once per 18 months~~ a frequency in accordance with the Surveillance Frequency Control Program.”

TS 5.5.13.c is revised as follows (deleted text in ~~strikeout~~ and added text in *italics*):

“Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested ~~every 31 days~~ based on ASTM D-2276, Method A-2 or A-3 at a frequency in accordance with the Surveillance Frequency Control Program;”

Entergy has determined that the relocation of the periodic frequencies associated with these specifications are consistent with the intent of TSTF-425, Revision 3, and with the NRC's model SE dated July 6, 2009 (74 FR 32001). The subject TS Section 5.5 frequencies are periodic frequencies and do not meet the scope exclusion criteria identified in Section 1.0, "Introduction," of the model SE. These changes are similar to the SR frequency relocation (i.e., TS 6.5.17.d frequency) that the NRC approved for Waterford Unit 3 in License Amendment 249 (ML16159A419) as described in Item 7 herein.

In accordance with TSTF-425, changes to the frequencies for these surveillances would be controlled under the SFCP. The SFCP provides the necessary administrative controls to require that surveillances related to testing, calibration and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Changes to frequencies in the SFCP would be evaluated using the NRC approved methodology and probabilistic risk guidelines contained in NEI 04-10, Revision 1.

5. NRC letter dated April 14, 2010 (ML100990099), provides a change to an optional insert (Insert #2) to the existing TS Bases to facilitate adoption of the Traveler. The TSTF-425 TS Bases insert states the following:

“The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.”

This statement only applies to frequencies that have been changed in accordance with the SFCP and does not apply to frequencies that are relocated but not changed. Consistent with NUREG-1430, Revision 4 (ML12100A177), Entergy has replaced the TSTF-425 TS Bases Insert #2 with the following variations based on the context of the specific Bases:

Individual SR Bases –

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

Multiple SR Bases –

“The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.”

SR Bases containing additional frequencies not controlled in the SFCP –

“The periodic Surveillance Frequency[ies] is[are] controlled under the Surveillance Frequency Control Program.”

6. Due to the relocation of SR frequencies and replacing the frequencies with the statement “In accordance with the Surveillance Frequency Control Program,” there are multiple SRs that moved to the next page as identified in the markup of the TS pages (Attachment 3). The following new pages are added because of SRs moving to the next page:

Page 3.3.1-5 (SR 3.3.1.5 moved to Page 3.3.1-4 and Table 3.3.1-1 moved to new Page 3.3.1-5)

Page 3.11-4 (SR Notes moved to Page 3.3.11-3 and Table 3.3.11-1 moved to new Page 3.3.11-4)

Page 3.6.6-2 (SRs 3.6.6.3 and 3.6.6.4 moved to new Page 3.6.6-2)

Page 3.8.1-7 (SR 3.8.1.9 moved to new Page 3.8.1-7)

Page 3.8.6-4 (SR 3.8.6.6 moved to new Page 3.8.6-4)

These changes (SRs moving to the next page) are administrative changes with no impact on the NRC's model SE dated July 6, 2009 (74 FR 32001).

7. TS 5.5.5.d is revised as follows (deleted text in ~~strikeout~~ and added text in *italics*):

Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS in accordance with the Surveillance Frequency Control Program. The results shall be trended and used as part of the 18 month assessment of the CRE boundary assessment specified in TS 5.5.5.c.

TSTF-425 includes the relocation of the frequency for NUREG 1430, SR 3.7.10.4, associated with verifying one Control Room Emergency Ventilation System (CREVS) train can maintain a positive pressure relative to adjacent area(s). This SR was revised under TSTF-448, “Control Room Habitability,” to perform control room envelope unfiltered air leakage testing in accordance with the Control Room Envelope

Habitability Program. The requirement to perform the relative pressure surveillance was included in the new NUREG 1430, TS 5.5.18, "Control Room Envelope (CRE) Habitability Program," as TS 5.5.18.d. ANO-1 adopted TSTF-448 in Amendment 239 dated October 2009 (ML082540799), designating the Control Room Envelope Habitability Program as TS 5.5.5 with the subject surveillance requirement as TS 5.5.5.d. Therefore, the frequency change identified for NUREG-1430 SR 3.7.10.4 in TSTF-425 is being adopted as the ANO-1 TS 5.5.5.d frequency. This is an administrative deviation from TSTF-425 with no impact on the NRC staff's model SE dated July 6, 2009 (74 FR 31996). In addition, on July 26, 2016, the NRC approved a similar SR frequency relocation (i.e., TS 6.5.17.d frequency) in the Waterford Unit 3 TSTF-425 License Amendment 249 (ML16159A419).

8. Entergy proposes to relocate surveillance frequencies with periodicities different from those in TSTF-425 (e.g., SR 3.3.1.2). These differences have been approved by the NRC in prior ANO-1 amendment requests and are an administrative deviation from TSTF-425 with no impact on the NRC staff's model SE dated July 6, 2009 (74 FR 31996).

Entergy proposes to relocate surveillance frequencies, except those that reference other approved programs, that are purely event-driven, are event-driven but have a time component for performing the surveillance on a one-time basis once the event occurs, or are related to specific conditions. Entergy considers the differences listed herein to be minor variations or deviations of the type permitted by TSTF-425.

Attachment 7 provides a cross-reference between ANO-1 TS SRs and the NUREG-1430 SRs included in TSTF-425. This attachment includes a summary description of the referenced TSTF-425/ANO-1 TS SRs, which is being provided for information purposes only and is not intended to be a verbatim description of the TS SRs. This cross-reference highlights the following:

- NUREG-1430 SRs included in TSTF-425 and corresponding ANO-1 TS SRs with plant-specific surveillance numbers;
- NUREG-1430 SRs not included in TSTF-425 that meet TSTF criteria for frequency relocation and corresponding ANO-1 TS surveillances with plant-specific surveillance numbers, as applicable;
- NUREG-1430 SRs included in TSTF-425 that are not contained in the ANO-1 TS; and
- ANO-1 plant-specific TS surveillances that meet TSTF criteria for frequency relocation but are not contained in either NUREG-1430 or markups in TSTF-425.

Inclusion of Attachment 7 is provided to assist the NRC staff's review of the proposed amendment and has no impact on the NRC staff's model SE dated July 6, 2009 (74 FR 32001).

3.0 Regulatory Analysis

3.1 Applicable Regulatory Requirements

A description of the proposed changes and their relationship to applicable regulatory requirements is provided in TSTF-425, Revision 3 and the NRC's model safety evaluation published in the Notice of Availability dated July 6, 2009 (74 FR 31996). Entergy has concluded that the relationship of the proposed changes to the applicable regulatory requirements presented in the Federal Register notice is applicable to ANO-1.

3.2 No Significant Hazards Consideration

Entergy Operations, Inc. (Entergy) has reviewed the proposed no significant hazards consideration determination (NSHC) published in Federal Register 74 FR 32000, dated July 6, 2009. Entergy has concluded that the proposed NSHC presented in the Federal Register notice is applicable to Arkansas Nuclear One, Unit 1 and is provided as Attachment 6 to this amendment request, which satisfies the requirements of 10 CFR 50.91(a).

3.3 Precedent

Relocation of surveillance frequencies to a licensee controlled program was approved for multiple licensees including; Cooper Nuclear Station per License Amendment No. 258 issued on March 31, 2017 (ML17061A050), D.C. Cook Units 1 and 2 per License Amendment Nos. 334 and 316, respectively, issued on March 31, 2017 (ML17045A150), and Brunswick Units 1 and 2 per License Amendment Nos. 276 and 304, respectively, issued on May 24, 2017 (ML17096A129).

3.4 Conclusion

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

4.0 Environmental Consideration

Entergy has reviewed the environmental consideration included in the NRC's model safety evaluation published in the Federal Register on July 6, 2009 (74 FR 32006). Entergy has concluded that the NRC's findings presented therein are applicable to Arkansas Nuclear One, Unit 1, and the determination is hereby incorporated by reference for this application.

ATTACHMENT 2

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DOCUMENTATION OF PRA TECHNICAL ADEQUACY

DOCUMENTATION OF PRA TECHNICAL ADEQUACY

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1. PURPOSE

The purpose of this report is to document the technical adequacy of the Arkansas Nuclear One, Unit 1 (ANO-1) Probabilistic Risk Assessment (PRA) model to support the implementation of the Surveillance Frequency Control Program (SFCP), also referred to as Technical Specifications Initiative 5b (Reference 1). ANO-1 intends to follow the guidance provided in NEI 04-10, Revision 1 (Reference 2), in evaluating proposed surveillance test interval (STI) changes (also referred to as “surveillance frequency” changes).

2. SCOPE

As explained in NEI 04-10, the Technical Specifications Initiative 5b uses a risk-informed, performance based approach for establishment of the surveillance frequencies, where PRA methods are used to determine the risk impact of the revised intervals. The PRA technical adequacy is addressed through NRC Regulatory Guide (RG) 1.200 (Reference 3), which references the ASME/ANS PRA standard, RA-Sa-2009 (Reference 4), for internal events at power. Risk impacts associated with fire, seismic, external events and shutdown activities may be considered quantitatively or qualitatively.

NEI 04-10 guidance includes the five key safety principles described in RG 1.174 (Reference 5), which are followed as part of this risk-informed Technical Specification Interval change program. The five key safety principles are:

1. Change meets current regulations unless it is explicitly related to a requested exemption or rule change
2. Change is consistent with defense-in-depth philosophy
3. Maintain sufficient safety margins
4. Proposed increases in core damage frequency (CDF) or risk are small and consistent with the Commission’s Safety Goal Policy Statement
5. Use performance-measurement strategies to monitor the change

The ANO-1 PRA model Revision 5p0, as developed for the Equipment Out of Service (EOOS) and Mitigating System Performance Index (MSPI) applications, is the current model of record used at ANO-1 for at-power, internal events. This model and its technical content was constructed and documented to meet the ASME/ANS PRA standard (Reference 4).

The ANO-1 fire PRA model update was completed in 2016. This model was constructed to meet the requirements of NUREG/CR-6850 (References 9 and 10). The PRA model quantification methodology used at Entergy Operations, Inc. (Entergy), nuclear sites is common and well-known to the industry.

Entergy’s approach for maintaining, updating and documenting the PRA models at all Entergy nuclear sites is controlled in the fleet procedures. These procedures are consistent with the guidance of the ASME/ANS PRA standard (Reference 4). The procedural process is comprehensive and detailed, which in turn provides the basis for establishing and maintaining the technical adequacy of the models, as well as ensuring the models reflect the as-built, as-operated plant configuration of the sites. In addition, self-assessments and independent peer reviews are also utilized by Entergy, which reassure the confidence in the approach and overall adequacy of the models against the recognized industry standards and methodologies.

Sections 2.1 and 2.2 describe the general change process and PRA adequacy requirements, respectively, required to support the Initiative 5b. Section 3 documents the technical adequacy of the ANO-1 PRA model specifically.

2.1 Surveillance Frequency Change Process

NEI 04-10 describes the required steps to be followed to adjust an STI. A summary is presented below.

- Once the STI requiring adjustment is selected, NRC regulatory commitments are collected and reviewed. If any prohibitive commitments are identified, such are examined to determine if the commitment can be changed. If there are no prohibitive commitments, or the commitments may be changed using a commitment change process based on NRC endorsed guidance, then evaluation of the STI revision proceeds. If a regulatory commitment exists and the commitment change process does not permit the change, then the STI revision is not implemented (NEI 04-10, Steps 0-4 (Reference 2)).
- The PRA technical adequacy is evaluated using guidance from RG 1.200 (Reference 3). The RG addresses the need to evaluate important assumptions that relate to key modeling uncertainties (such as reactor coolant pump seal models, common cause failure methods, success path determinations, human reliability assumptions, etc.). Further, the RG addresses the need to evaluate parameter uncertainties and demonstrate that calculated risk metrics (i.e., CDF and large early release frequency (LERF)) represent mean values. The identified “gaps” to Capability Category (CC) II requirements from the endorsed PRA standards in the RG and the identified key sources of uncertainty serve as inputs to identifying appropriate sensitivity cases (NEI 04-10, Step 5 (Reference 2)).
- Select the revised STI value and revise any changes to the test strategy (NEI 04-10, Step 6 (Reference 2))
- Qualitative considerations or qualitative analyses are developed for the STI revision. Qualitative considerations include surveillance test and performance history, past industry and plant-specific experience, impact on defense-in-depth protection, among other considerations (NEI 04-10, Step 7 (Reference 2))
- Perform quantitative and/or qualitative PRA assessments. Steps 8 through 12 in NEI 04-10 provide details regarding the use of PRA for evaluating the STI. The use of the PRA includes: determining if the structures, systems and components (SSCs) in question are modeled in the PRA, whether the SSCs or operator actions can be modeled (and make changes to the model if possible) or not, perform qualitative assessments as needed, evaluate total and cumulative effect on CDF and LERF, and perform sensitivity studies as needed.
- The results and proposed STI changes are documented and summarized for consideration by the Integrated Decision-making Panel (IDP). The IDP is usually comprised of the site Maintenance Rule expert panel, a surveillance test coordinator, and a subject matter expert. The IDP approves or rejects the STI changes (with the possibility of adjustments if applicable). If the IDP approves the STI changes, these are documented and implemented. The IDP is also responsible for reviewing the performance monitoring results and providing feedback, if the STI changes, once implemented, result in unsatisfactory performance (NEI 04-10, Steps 16-20 (Reference 2)).

2.2 Technical Adequacy of a PRA

As previously discussed, NEI 04-10 endorses the guidance of the NRC Regulatory Guide 1.200 (Reference 3) for the PRA technical adequacy determination. For the purposes of this report, Section 4.2 of RG 1.200 is used in support of Initiative 5b licensee applications. It is important to note that the scope of Initiative 5b applications is broad, and PRA assessments needed for each application vary from application to application. The following requirements are noted in Section 4.2 as necessary to demonstrate that the technical adequacy of the PRA is of sufficient quality to support the application submittal:

1. To address the need for the PRA model to represent the as-designed or as-built, as-operated plant.
2. Identification of permanent plant changes (such as design or operational practices) that have an impact on those SSCs modeled in the PRA, but have not been incorporated in the baseline PRA model. If a plant change has not been incorporated in the PRA, the licensee provides a justification of why the change does not impact the PRA results used to support the application. This justification should be in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application decision were not adversely impacted (remained the same).
3. Documentation that the parts of the PRA required to produce the results used in the decision are performed consistently with the standard as endorsed in the appendices of the RG. If a requirement of the standard (as endorsed in the appendix to the RG) has not been met, the licensee is to provide a justification of why it is acceptable that the requirement has not been met. This justification should be in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application were not impacted (remained the same).
4. A summary of the risk assessment methodology used to assess the risk of the application, including how the base PRA model was modified to appropriately model the risk impact of the application and results (note that this is the same as that required in the application-specific regulatory guides).
5. Identification of the key assumptions and approximations relevant to the results used in the decision-making process. Also, include the peer reviewers' assessment of those assumptions. These assessments provide information to the NRC staff in their determination of whether the use of these assumptions and approximations is appropriate for the application, or whether sensitivity studies performed to support the decision are appropriate.
6. A discussion of the resolution of the peer review (or self-assessment, for peer reviews performed using the criteria in NEI 00-02) facts and observations that are applicable to the parts of the PRA required for the application. This discussion should take the following forms:
 - a discussion of how the PRA model has been changed,
 - a justification in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application decision were not adversely impacted (remained the same) by the particular issue.

7. The standards or peer review process documents may recognize different capability categories or grades that are related to level of detail, degree of plant specificity, and degree of realism. The licensee's documentation is to identify the use of the parts of the PRA that conform to capability categories or grades lower than deemed required for the given application (Section 1-3 of ASME/ANS RA-Sa-2009).

This PRA technical adequacy report addresses the quality of the PRA to support relocation of STI frequencies to a licensee controlled document. There are no STI changes proposed for this Initiative 5b application. Items 3 and 4, above, are addressed when preparing an STI change request and are, therefore, not covered in this report. The remaining items above are discussed in Section 3.

3. ANO-1 PRA TECHNICAL ADEQUACY

3.1 Discussion

The ANO-1 PRA models are controlled in accordance with Entergy procedures consistent with the requirements provided in the ASME/ANS PRA Standard, as previously stated in Section 2. Entergy procedures define the process to be followed to implement scheduled and interim PRA model updates and to control the PRA model files. In addition, the procedure also defines the process for identifying, tracking, and implementing model changes, and for identifying and tracking model improvements or potential issues that may affect the model. Model changes that are identified are tracked via model change requests (MCRs), which are entered in the Entergy MCR database.

Periodic PRA model updates are typically performed at least once every four years, with the option of extending the frequency for up to two years, such that the total update period does not exceed six years. Extensions are justified showing that the PRA model continues to adequately represent the as-built, as-operated plant, and must be approved by management.

The ANO-1 PRA model 5p0 was approved in 2016. The internal flood model upgrade was developed in 2016, underwent a focused-scope peer review in early 2017, and unresolved facts and observations (F&Os) are currently being addressed. Both models follow the guidelines of RG 1.200. Section 3.2 discusses the requirements in RG 1.200 to demonstrate PRA technical adequacy, as applicable to the current ANO-1 internal events and internal flooding models.

An ANO-1 fire PRA model (Reference 10) update was completed in 2016 and was developed in accordance with NUREG/CR-6850. Section 3.3 discusses the requirements in RG 1.200 to demonstrate PRA technical adequacy, as applicable to the current ANO-1 fire PRA model.

3.2 ANO-1 Internal Events and Internal Flooding PRA Model

3.2.1 Plant Changes Not Yet Incorporated

As discussed in Section 3.1, an MCR database tracks PRA issues or improvements identified by PRA personnel. The MCR database includes the identification of plant changes that could impact the PRA model.

As part of the PRA evaluation for each STI change request, sensitivity cases are expected to be explored for areas of uncertainty associated with unresolved items (peer review Findings for ASME/ANS PRA Standard CC II or plant changes) that would impact the results of the STI change evaluation, prior to presenting the results of the risk analysis to the IDP.

A review of the MCR database for ANO-1 identified several plant changes not yet implemented which may potentially impact the PRA. These are listed in Table 1, along with the engineering changes (ECs) and corresponding MCR numbers, as well as the potential impact on the PRA model.

Table 1
Summary of Plant Changes Not Yet Incorporated in the ANO-1 PRA

MCR Number	EC or Procedure	Description of Change	Importance to Application
A1-3050	EC-3069	Currently ANO-1 recirculation lines for BWST through SFP purification loop are not seismically qualified. EC-3069 incorporates two valves CV-1438 and CV-1441 into line that will get a close signal on ESAS channel 1 and 2 respectively. If a seismic event occurs, operations will be required to close these valves per the procedure. The current estimation is that it will take the operations approximately 30 minutes to take the action to close the valves.	The modification is intended for seismic considerations. The potential impact will be addressed by STI change evaluations performed in accordance with the SFCP.
A1-4837	EC 41875	<p>The NFPA-805 circuit modification described in EC41875 prevents spurious operation of the valves following fire impact to the associated circuits regardless of where the cable is damaged. However, the FPRA was only adjusted to remove the fire impact to the valves in fire zone 129-F. The impact in other fire zones and fire areas has not been removed from the FPRA.</p> <p>The circuits being modified are for AOVs CV-1052, CV-4400, CV-7401, CV-7402, CV-7403, and CV-7404 as well as MOVs CV-1053, CV-4446, CV-5611, and CV-5612.</p>	The modifications have been incorporated in the fire PRA model, but the MCR is not yet closed. Therefore, this plant change has been incorporated, and has no impact in STI change evaluations performed in accordance with the SFCP.
A1-4862	EC 44500	During the FPRA model revision (re-baseline) for the NFPA-805 LAR RAI responses, update the random failure probability for the new DC supplies to bound the modification described in EC44500. Also update the random failure probabilities assumed for the new AFW pump if necessary to reflect that modification.	The modification and updated data has been incorporated in the fire PRA model, but the MCR is not yet closed. Therefore, this plant change has been incorporated, and has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	EC or Procedure	Description of Change	Importance to Application
A1-4933	SDS-008 1202.008 1203.048	<p>B5b actions/equipment can potentially be credited in the model to provide an alternate feedwater and RWT/BWST makeup from the B5b portable fire pump. By having an additional pump to EFW/AFW for long term scenarios can provide some benefit for those scenarios.</p> <p>Also, procedure SDS-008 Change 002 contains the following statement that could add additional actions in the PRA: "IF Unit 2 has AC power, THEN consider performing Security Event (1203.048), Attachment G, Cross-Tying 4160V Buses Between Units."</p> <p>Blackout Procedure 1202.008 Rev. 15 provides a reference to 1203.048 for supplying power from Unit 2 to Unit 1 via 2A9.</p> <p>Once procedure 1203.048 is reviewed, there may be additional actions that can provide enhancement to the PRA model.</p>	<p>The modification involves the retrieval of additional equipment from storage and alignment for mitigation of certain accident scenarios. The potential impact will be addressed by STI change evaluations performed in accordance with the SFCP.</p>
A1-5051	EC 44044 and child ECs	<p>EC 44044 and child ECs 46389, 46390, and 46391 make up the electrical portion of ANO-1's BDBEE (Beyond-Design Basis External Event) plan. The scope of this project includes installing new breakers into the existing Load Center B-5 and Motor Control Centers B-55, B-61, and B-65, and the permanent raceway, cables, and termination panels required to use them during a BDBEE. Portable diesel generators will be stored outside of the protected area (Ref. EC-44045 "ANO Flex Storage Building") at separate locations to assure availability of at least one post BDBEE. A portable generator will be moved into a position at the common PASS (Post Accident Sampling System) building on the west side of the auxiliary building between the ANO-1 BWST (Borated Water Storage Tank) and the ANO-2 RWT (Refueling Water Tank). Once at this location it will be connected to the permanently installed infrastructure previously described.</p>	<p>The modification involves the retrieval of additional equipment from storage and alignment for mitigation of certain accident scenarios. The potential impact will be addressed by STI change evaluations performed in accordance with the SFCP.</p>
A1-5821	1106.006	<p>Reviewing procedure 1106.006 Rev. 100 (dated 1/5/2017), a change was made in section 8.3 for EFW pump suction transfer to transfer to Service Water instead of the CST. This change removes the allowance to transfer to the CST. This change was based on the SW supply being judged as the preferred for transfer vice CST T-41 since it is the safety-related water supply and it is currently unknown when vortexing might occur when aligned to the CST. However, the current revision (dated 4/6/2016) of procedure 1203.012K Rev. 48, still maintains the CST (T-41) as an option with T-41B empty when P7A/P7B suction pressure is low, but references 1106.006 that does not credit T-41 in the EFW pump suction transfer section. The QCST T-41B, CST T-41, and Service Water are credited in the PRA to provide flow to EFW.</p>	<p>The modification involves use of alternate system alignments for mitigation of certain accident scenarios. The potential impact will be addressed by STI change evaluations performed in accordance with the SFCP.</p>

3.2.2 Peer Review Facts and Observations (F&Os)

The ANO-1 PRA model has undergone several peer reviews and self-assessments which document the model quality and identify any areas with potential for improvement. The following assessments for PRA quality have been performed and documented for the ANO-1 model:

- An industry peer review of the ANO-1 probabilistic safety assessment (PSA) model Revision 2p2 was conducted by the Babcock and Wilcox Owners Group in 2002 (Reference 6). The peer review concluded that there were several areas where the ANO-1 model needed improvement. The ANO-1 PSA model update Rev 3p0 completed in August 2006 addressed the Finding-level F&Os from this peer review.
- In preparation for ANO-1's transition to National Fire Protection Association (NFPA) 805 standard, a gap assessment of the ANO-1 PSA 3p0 internal events PRA model was completed. The gaps impacting the fire PRA were closed to meet the NFPA transition schedule. The ANO-1 Internal Events PSA model was updated (Rev 4p0 completed in 2009) to meet the RG 1.200, Revision 1, standards.
- In August 2009, a peer review of the Rev 4p0 ANO-1 PSA Model was performed and documented in the Peer Review Report (Reference 7). This peer review documented eighty-six (86) new F&Os including forty-three (43) Findings, forty-two (42) Suggestions, and one (1) Best Practice. Most of the findings pertained to documentation issues. The conclusion of the review was that the ANO-1 PRA substantially met the ASME PRA standard at CC II or better, except for those LERF SRs reviewed against CC-I.
- In September 2017, a self-assessment of all ANO-1 PRA models (internal events, external, and shutdown) was performed. Several gaps were identified and are being tracked in the Entergy's Paperless Condition Reporting System (PCRS).

The ANO-1 internal events model revision 5p0 was approved in 2016. This is the current PRA model as stated in Section 2. The peer review findings and the associated resolutions (with the exception of the 12 Findings superseded by the internal flooding focused scope peer review described below), as well as the remaining unresolved findings related to the internal events PRA model, are documented in the MCR database and are presented in Table 2.

The current ANO-1 internal flooding model was developed in 2016, and a focused scope peer review was completed in February 2017 against the current ASME/ANS PRA standard and RG 1.200. The results are detailed in report ENTGANO150-REPT-001 (ENERCON report) (Reference 8). The results of the assessment show that approximately 90% of the internal flooding Supporting Requirements (SRs) were met at the CC II level. Twenty-four (24) F&Os were issued during this peer review, including twelve (12) Findings and twelve (12) Suggestions. The F&Os were documented in the MCR database to be resolved. The unresolved F&Os related to internal flooding, as documented in the MCR database, are judged to have minimal or no impact on the overall results. The Findings related to the internal flooding PRA model are also reported in Table 2. An update to the current internal flooding model is in progress to address the unresolved findings and to incorporate model refinements.

3.2.3 Consistency with Applicable PRA Standards

The ANO-1 PRA model revision 5p0 was assessed to meet the ASME/ANS PRA standard (Reference 4) CC-II of the SRs, except where noted in Table 3. Current Entergy PRA documentation includes an individual self-assessment that documents how each high-level requirement (HLR) and SR are met. A gap in documentation related to this individual self-assessment was identified and documented in MCR A1-5584 (missing table from Human Reliability Analysis and Quantification packages). This documentation issue is expected to be resolved as part of the model update.

The latest full-scope peer review for ANO-1 was conducted in August 2009 using RG 1.200, Rev 2. Since then, model revision 5p0 was completed which ensured all the significant F&Os from the peer review were addressed. All the F&Os are captured and documented in the MCR database. A search of ANO-1 MCRs related to peer review F&Os was performed. Finding-level F&O MCRs related to the internal event and internal flooding models are listed in Table 2, along with their disposition/resolution, if resolved, and the impact on the application.

Table 2

List of Finding F&Os against the ANO-1 Internal Events and Internal Flooding Models

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3873	Resolved	QU-B1	<p>Reviewed PRA-A1-01-001S02 Rev 1: Method specific limitations and features that could impact results are not identified.</p> <p>Possible Resolution: Identify and document the limitations and features of the methodology that could impact the PRA results.</p>	Section 5.4 of the Quantification notebook (PSA-ANO1-01-QU ANO-1 Integration and Quantification Work Package) documents the 'Limitations of the Quantification Methodology Results which includes the method specific limitations that could impact results (consistent with SR QU-B1).	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3876	Resolved	SY-A4	<p>PRA-A1-01-001S11, Rev 1., Section 4.0 documents the plant walkdowns and system engineer discussions for each system. The system engineer discussion is part of the latter. There is no indication in the walkdown/ discussion documentation that the modeling was verified to represent the as-built, as-operated plant.</p> <p>Possible Resolution: Although it is acknowledged that system walkdowns and discussions with system engineers have been conducted and that spatial and environmental hazards were identified; the documentation of these activities does not convey that the model indeed does represent the as-built/as-operated plant. Perform/document additional walkdowns and/or discussions focusing on confirmation that the model represents the as-built, as-operated plant, or document the satisfaction of this requirement if the existing walkdowns and/or discussions have already accomplished this goal.</p>	<p>Discussion was added to Section 4.0 (SYSTEM WALKDOWNS AND INTERVIEWS) of the main body of the system notebook engineering report that describes the process that ensures the PRA model reflects the ABAO plant. Namely, the PRA model has been used in day-to-day plant operations (via EOOS Monitor) and other processes, such as Maintenance Rule and other plant applications, for over 15 years. Any errors identified through these applications are documented in either the Condition Reporting system or the Model Change Request database. Also, EN-DC-151 requires review of plant and procedure changes to ensure that changes that affect the PRA are addressed as part of the model update. Section 2.1.6 (System Engineering Interviews) was added to each notebook that refers back to the main body discussion and the individual system engineer interviews that are currently documented in the walkdown checklists.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3881	Resolved	AS-B3, SY-A22, SY-B14	Reviewed PRA-A1-01-001S03-EC13903: There is no purposeful description of the phenomenological conditions associated with each accident sequence, as required by the supporting requirement. Possible Resolution: Include a description of the phenomenological conditions for each sequence.	Additional discussions including phenomenological conditions have been documented in the updated ANO-1 ATWS notebook (PSA-ANO1-01-AS-01). The updated report contains phenomenological discussion in the introduction section to each sequence category.	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.
A1-3884	Resolved	QU-D6	The quantification approach which includes modularization of the IE fault trees precludes calculation of importances for events within the modules. Possible Resolution: Provide discussion or tabulation of significant contributors to CDF from IEs as well as from mitigating systems.	(Section 4.6 of the Quantification notebook (PSA-ANO1-01-QU ANO-1 Integration and Quantification Work Package) documents the application of 'modules'. The second table in the section lists the significant contributors to each IE fault tree module (along with % contribution).	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.
A1-3885	Resolved	QU-E4	This uncertainty 'characterization' has not been performed. Possible Resolution: Perform a characterization of the sources of uncertainty. It is recommended that the EPRI 1016737/NUREG-1855 approach be applied.	Updated report PSA-A1-ANO-01-QU-01 (Sensitivity and Uncertainty) identifies many possible sensitivity cases and quantified most of these cases and qualitatively assessed those that could not be quantified. The examination of uncertainty in PSA-A1-ANO-01-QU-01 is consistent with EPRI 1016737/NUREG 1855 methodology. Parametric uncertainty is calculated for CDF and LERF results with uncertainty distributions and results for mean, 5%, 50%, 95% standard deviation provided (CDF and LERF).	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3885 (continued)				The PSA-A1-ANO-01-QU-01 appendix evaluated potential uncertainty from many of the model elements and qualitatively discusses the impact or why it is not a covered for each.	
A1-3886	Resolved	QU-D7	<p>Reviewed PRA-A1-01-001 Rev. 1 and EN-NE-G-014. The guideline instructs that reviews include a comparison of the basic event risk importances and system importances in the current model to the previous model. This appears to address the BE review question but there is no explicit discussion of the BE review.</p> <p>Possible Resolution: It would be helpful for the BE importances to be tabulated to demonstrate that the BEs had been reviewed.</p>	Appendix R and Appendix S in the PSA-A1-ANO-01-QU document (ANO-1 Integration and Quantification Work Package) provide a comparison of update 5p0 results to the previous revision 4p0 results. This includes comparison of the top cutsets (and details on why items have gone up or down) - Appendix R. Appendix S provides the results by Sequence including a comparison of the % contribution of each sequence for both the old and updated models. The table also provides insights into what has changed to impact the change in results.	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.
A1-3890	Resolved	LE-G5	<p>Reviewed PRA-A1-01-001S12, Revision 1. Section 2.1 identifies several limitations of the applicability. However, the noted limitations do not address technical limitations that might impact the use in applications.</p> <p>Possible Resolution: Document the limitations of the technical aspects of the analysis.</p>	Section 2.1 of the updated LERF analysis, PSA-ANO1-01-LE, Rev. 0, details the limitations including technical limitations and the potential for them to impact use in applications.	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3893	Unresolved	IE-D1	<p>Some basic events (e.g., XMP119BBAF) applied in the calculation of IE frequencies developed for plant-specific fault trees have used calculation method 3 in CAFTA. The use of calculation method 3 ($1-e^{-\lambda t}$) produces a probability (always < 1) rather than a frequency which can be greater than 1. Calculation method 1 (λt) in CAFTA should be used for those basic events whose result is intended to be a frequency of failure, not a probability of failure.</p> <p>A discussion of the use of this calculation method is not provided, although, during discussion of this issue, the PRA staff indicated that the limitations of the selected approach were understood.</p> <p>Possible Resolution: Provide a description of the approach taken for calculation of the basic event values within support system initiating event fault trees, and include the limitations of the approach.</p>	<p>The initiating event fault trees (IEFTs) have been revised extensively during the Revision 5 update. All events in the IEFT logic have the correct calculation type and the fault tree result is a frequency (yearly frequency). The Instrument Air (IA), Power Conversion System (PCS), Service Water (SW), and Intermediate Cooling Water (ICW) are the systems with IEFT logic and each system notebook has an attachment documenting the IEFT development (including how event data is developed for each modeled event). The quantification notebook (PSA-ANO1-01-QU) also documents how the IEFTs are quantified and how the frequencies are linked to the model.</p>	<p>Additional information on the IEFT development was added to the system notebooks for the model revision 5 update. This is a documentation issue and additional discussion of flooding results is not expected to impact STI evaluations performed in accordance with the SFCP.</p>
A1-3894	Resolved	IE-C5	<p>Initiating event frequencies are not calculated in the manner required by the IE-C5. The IE units are in critical years versus reactor (calendar) years.</p> <p>Possible Resolution: Calculate the frequencies in the units specified by SR IE-C5.</p>	<p>The results in the Quantification notebook (PSA-ANO1-01-QU ANO-1 Integration and Quantification Work Package) are presented in both "per reactor critical year" and "per reactor year. Also ensured IE frequencies are calculated per IE-C5.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3897	Unresolved	IE-C8, QU-D6	<p>Section 5.3 of PRA-A1-01-001S06, Revision 2 identifies those initiating events that are quantified by means of a plant specific fault tree.</p> <p>Appendices C, D, E, and F provide additional detail on each of the 4 modeled initiating events.</p> <p>Per Appendices E and F, the PSA logic model is used as the starting point for the IE model; however, a number of modeling simplifications are made as identified in Appendix C. These simplifications may cause the model to fall out of compliance with the SY requirements.</p> <p>Possible Resolution: Use the system fault tree with necessary data (exposure time) changes to evaluate IEs.</p>	<p>The initiating event fault trees (IEFTs) have been revised extensively during the Revision 5 update. The base model fault tree logic is the starting point for the IEFTs. The revised models are more thorough than past models and fully meet the PRA Standard (including element IE-D1). IEFT development is documented in Attachments to the individual system notebooks Instrument Air (IA), Power Conversion System (PCS), Service Water (SW), and Intermediate Cooling Water (ICW) system notebooks (PSA-ANO1-01-SY_R0)</p>	<p>The IEFTs were revised during the revision 5 update to be more detailed and meet the PRA standard. A very small change to the results is expected. Impact of this finding is expected to be assessed in a case-by-case STI evaluations.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3902	Resolved	LE-A4, LE-C7, LE-E1	<p>Direct linkage of the CDF sequences into the LERF tree assures that system level dependencies (e.g., power and cooling water) can be accounted for.</p> <p>No HRA for human actions is provided.</p> <p>Possible Resolution: Develop human error probabilities using the ANO-1 selected HRA process.</p> <p>Ideally, the human and hardware components of Split Fraction for No Intentional or Unintentional RCS Depress pre I-SGTR for Non-SBOs should be separated. A JHEP analysis can then be included in the result.</p>	<p>Logic uses direct linkage of CDF sequences into LERF (system level dependencies are accounted for).</p> <p>Any LERF specific HEPs contained in the fault tree logic were developed using the same HRA process as other human events. Actions that are included as part of the split fractions used in LERF have no dependency with actions in the CDF model, or are not credited. For example, the action for the Operator to use the ADV to minimize MSSV demands is assumed to fail.</p> <p>The action to “bump the pumps” to clear the RCS loop seals may actually be detrimental, so assuming a high probability of success is conservative. If the operators do not perform this action or fail at the action, the LERF is actually decreased. Thus, dependency of this action with existing HFEs does not make sense.</p> <p>Recovery of offsite power prior to vessel failure for LERF is modeled in a similar way as recovery of offsite power prior to core damage for CDF. Any dependencies are already accounted for in the Recovery Rules file.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3903	Resolved	DA-D4	<p>Reviewed PRA-A1-01-001S05_EC15022. Section 3.1.5 provides guidance on checking results for those components where no plant failures have occurred. If the check (operating hours > 0.5*MTTF) fails, the generic data is used. The intent is to guard against undue influence of not statistically significant plant data.</p> <p>Guidance to check results is provided in CE-P-05.07, Rev. 01. Some posterior distributions do not appear reasonable (e.g., RYT P1, T7F D1)</p> <p>Possible Resolution: Provide additional examples in the guidance of what constitutes "unreasonable" to improve the confidence of detection and correction.</p>	<p>The use of generic and plant specific data and the application of reasonable 'priors' has been addressed in the updated data analysis for the revision 5 model update. The updated documentation and application of generic data (given no relevant plant specific failures) meets the requirements of Standard element DA-D4 (same element name/number for old and updated Standard). All posterior distributions judged to be reasonable through both vendor and Entergy review. PSA-ANO-01-DA-01 Revision 0 contains the relevant analysis on plant specific data and the application of Bayesian updating. The outputs from the Bayesian update (excel sheets) were reviewed to ensure the results are reasonable. The software was benchmarked against known Bayesian problems to ensure correct software operation.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3905	Resolved	QU-D4, IE-A1, IE-B3	<p>Reviewed cutset file A1_R4_1e-13_rec_M and fault tree A1R4P0EOS and PRA-A1-01-001, Revision 1, Section 6.4. This observation result from comparisons to similar plants.</p> <p>Spurious opening of SRV evaluation needs to be reexamined.</p> <p>Possible Resolution: Revisit the classification of this event as cc. Update the initiating event analysis with the proper classification of this event.</p>	<p>PSA-ANO1-01-IE, Rev 0 (ANO-1 Initiating Events Analysis Work Package), Appendix D Table 14 was revised to map NUREG/CR-5750 IE Category G4 (Stuck Open: Pressurizer PORV) to initiator %IORV. Previously, this IE category was incorrectly mapped to initiator %T2 in Table H-2 of PRA-A1-01-001S06, Revision 2. This change is consistent with "Possible Resolution" provided by the F&O.</p> <p>The %M frequency should be increased by $4.89\text{E-}06/\text{rcry}$ ($0.1 \times 4.89\text{E-}05/\text{rcry}$), i.e., from $5.07\text{E-}05/\text{rcry}$ to $5.56\text{E-}05/\text{rcry}$, to account for a spurious opening of an SRV to a full open position (10% of the time). No change in the value of %S, currently $5.77\text{E-}04/\text{rcry}$. Given that it was too late in the ANO-1 Model 5p00 update to revise the value of %M a sensitivity analysis was included in the 5p00 update to show the associated risk increase is very small (~0.3% increase in the baseline case CDF). The value for %M will be revised in the Rev. 6 update.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3906	Resolved	LE-F2	<p>Reviewed PRA-A1-01-001S12, Revision 1 and PRA-A1-01-001 Revision 1. No documentation of such a review is present in these documents.</p> <p>Sensitivity studies are performed for important inputs to the analysis.</p> <p>Reviewed PRA-A1-001-S02, Appendix J. No indication that the expert panel reviewed the LERF results.</p> <p>Possible Resolution: Include a review of the LERF contributors as part of the expert panel review. Document this review in the expert panel report.</p>	<p>Review of the LERF results were performed at various points during the model development process. Specifically, LERF cutsets were included in the package for the Expert Panel review meeting and LERF results were discussed in the slides. Insights from the meeting were incorporated into both the final Level 1 and Level 2 models.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>
A1-3907	Resolved	SC-A2, SC-B3, SC-B4, AS-A9	<p>MAAP 4.0.5 provides detailed core damage sequences.</p> <p>PRA-A1-01-0015S014-EC14882 Section 4.0 A of the Level 1 Success Criteria Notebook credits the use of MAAP for a LLOCA blowdown phase. Based on current MAAP guidance, MAAP should not be used to model the blowdown and reflood stages of a LLOCA. DBA codes should be used in this case. Following reflood, MAAP can be used for the remainder of the LLOCA.</p> <p>Possible Resolution: Use a DBA code to determine LLOCA success criteria during the blowdown and reflood stage.</p>	<p>The ANO-1 PRA Level 1 Success Criteria Notebook/ PRA-A1-01-001S014, Rev. 0 provides the success criteria and systems needed for the ANO-1 PRA. The MAAP model is not recommended for the analysis of LLOCA during the early blowdown phase and reflooding due to the over-prediction of water retained in the primary system. As a result, MAAP analyses were not used to model LLOCA during this phase of the accident sequence. Instead, a design basis accident code was used to support the LLOCA analysis and success criteria during this phase of the accident sequence.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3907 (continued)				<p>A LLOCA analysis was completed for the purpose of defining the limiting design requirements for the LPI flow (ANO-1 Large Break Loss of Coolant Accident, Rev. 3). The LLOCA analysis is based on the Acceptance Criteria for ECCS submitted as part of the original ANO-1 licensing basis and represented as a double ended cold leg split at the pump discharge with an area equivalent to twice the cross-sectional area of the pipe (8.55 ft²).</p> <p>The LLOCA analysis for the LPI with limiting design requirements will be updated (XA) or replace (U) the basis document currently shown the SC notebook.</p> <p>For the LLOCA including RCS Inventory Control (U) success criteria, a MAAP run was used to demonstrate CFT success criteria meets the acceptance criteria. The MAAP model used for this analysis included the limiting LPI design requirements as determined from the ANO-1 LLOCA analysis. The MAAP model was acceptable for use in evaluating the RCS Inventory Control (U), because the success criteria were associated with restoring water levels in the RCS well after core reflood was established.</p>	

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3907 (continued)				<p>MAAP does not have limitations within these regions of the accident sequence analysis.</p> <p>Updated Success Criteria Notebook (PSA-ANO1-01-AS):</p> <ul style="list-style-type: none"> • Section 2.0 – Computer codes were reviewed and updated to reflect codes used including LLOCA analysis. • Section 4.0 – 1.) Updated basis documents in A.1. and A.2; 2.) Add a note to A.1 to clarify that the limiting LPI flow requirements in the MAAP Case #10 (AX2) are consistent with the ANO-1 LLOCA analysis. • Section 7.0 – Replaced reference #4 with updated ANO-1 LLOCA analysis. 	

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3909	Resolved	SC-C3, QU-E1, AS-C3, DA-E3, HR-I3, LE-F3, SC-C3, SY-C3, LE-G4, QU-E4, QU-F4	<p>There is no discussion of identification of issues related to modeling uncertainty.</p> <p>Possible Resolution: Provide the identification of sources of modeling uncertainty. It is recommended that the process described in EPRI 1016737/NUREG-1855 be incorporated.</p>	<p>Discussion of potential modeling uncertainty issues is included in the Sensitivity and Uncertainty document. PSA-A1-ANO-01-QU-01 (Sensitivity and Uncertainty) identified many possible sensitivity cases and quantified most of these cases and qualitatively assessed those that could not be quantified. The examination of uncertainty in PSA-A1-ANO-01-QU-01 is consistent with EPRI 1016737/NUREG 1855 methodology. Parametric uncertainty is calculated for CDF and LERF results with uncertainty distributions and results for mean, 5%, 50%, 95% standard deviation provided (CDF and LERF). The PSA-A1-ANO-01-QU-01 appendix evaluated potential uncertainty from most of the model elements and qualitatively discusses the impact or why it is not a covered for each.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3910	Resolved	DA-C10	<p>There is no evidence that surveillance tests have been evaluated to determine if portions of the tests or sub-elements have additional successes that should or should not be counted when estimating operational demands.</p> <p>Possible Resolution: Perform an assessment of the sub-elements of all surveillance tests to obtain accurate operational demands to be used in the PRA data.</p>	Plant specific failure rates were developed based on operator logs which includes surveillance tests.	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.
A1-3911	Resolved	DA-C4, DA-E2	<p>The primary source of failure data is the Maintenance Rule Database. This database was used to screen the component failures to determine if the Maintenance Rule Functional Failures were also PRA-relevant failures. It is suggested that an additional source of data be reviewed to determine if a failure may have occurred that did not result in a Mrule Functional Failure. In addition, a suggestion is provided to include a discussion or tabular display of those failures that are excluded from the data.</p> <p>Possible Resolution: Perform a scrub or review of EPIX, Condition Reports, Issue Reports, and/or plant specific LERs to determine if there are any additional failures that should be considered in the PRA Data Update to supplement the Maintenance Rule Database functional failures.</p>	The operator logs used for plant data is attached to the Data Notebook (PSA-ANO1-01-DA-01).	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3913	Resolved	DA-C13	<p>Interviews with knowledgeable plant personnel are documented in various locations in the PRA Data Analysis in various assumptions and as sources of data to estimate run times, demands, etc. It is not clear that Outage UA has been excluded from the Mrule Data since some Mrule functions may be outage related. There is no consideration for alignment of the AAC during a dual unit SBO. In addition it appears that the SU2 transformer could be credited to support both trains on Unit 2, however it is assumed to be aligned to Unit 1.</p> <p>Possible Resolution: Model the AAC unavailability to support either unit in the event that both or either unit requires it for LOSP mitigation and provide a documented basis for the flag alignment settings used for the SU2 transformer.</p>	The outage times were considered in the development of plant specific data.	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3916 and A1-5377	Resolved	IE-A1	<p>Section 4.0 of PRA-A1-01-001S06, Revision 2 describes the process used to identify initiating events. This process considered generic events as well as initiating events modeled in similar plants (TMI and Oconee). It was noted that the loss of Service Water initiator is relatively low significance in the ANO-1 PRA model. %T8, %T9, and %T10 are identified as Loss of Running SW, Loss of SW Loop 1 and Loss of SW Loop 2 respectively. There are no initiators that represent a total loss of all 3 SW Pumps (including common cause of all three to fail). An initiator for Loss of Lake is included in the model but that does not account for SW pump failures. In similar NSSS designs, the loss of SW is a significant contributor to CDF. The process is further prescribed in Fleet engineering guide EN-NE-G-006.</p> <p>Possible Resolution: Include the total loss of SW in the ANO-1 PRA.</p>	<p>Review of the loss of Service Water initiator removed the loss of SW Train A and Train B initiators based on the fact that the cross-tie valves are normally open. The %T8 initiator is based on a total loss of SW and thus addresses this MCR.</p> <p>As discussed in the SW system notebook, Attachment 1, the success criteria for the loss of SW initiator is based on one of three SW pumps. (See PSA-ANO1-01-SY, Appendix 13, Attachment 1).</p> <p>The following is the updated information from MCR A1-5377.</p> <p>Added Loss of SW Loop initiators %T9 and %T10 and updated total loss of SW initiator %T8. Also corrected several SW modeling errors from other Rev. 5 changes, primarily associated with system flow diversion.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3917	Resolved	AS-A5	<p>Reviewed PRA-A1-01-001S03-EC13903: The development of the Event Trees appears to be consistent with the plant design/operation. An isolated model error was found related to placement of an Human Action to cooldown and depressurize during a Steam Generator Tube Rupture. The error results in no account for HEP probability to fail to initiate the cooldown process high enough up in the SGTR event tree. A HEP should be place near the top gate to yield simple sequences where an SGTR occurs, no equipment failures occur but the operators fail to cooldown and depressurize.</p> <p>Possible Resolution: Add a HEP to the SGTR sequence model high enough in the model logic to verify that OPS successfully initiates the SGTR cooldown.</p>	The model includes HFE LHF1RCSDHP, which has been renamed as LPI-HFC-FO-RCSDH, which addresses Failure to Cooldown RCS and Isolate Break (SGTR) with DHR System.	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3919	Resolved	IE-C3	<p>In Table 3 of PA-A1-001S06, Rev. 2, where screened potential initiating events are considered, two human recovery actions are used to justify not modeling an event as an initiator (i.e., Steam line break and HPI actuation). However, no justification (i.e., training or procedures) was provided.</p> <p>Possible Resolution: Provide the appropriate training documents or procedures showing these particular human actions.</p>	<p>The sentences in Table 3 of PRA-A1-01-001S06 describing operator response for the Steam Line Break initiator were deleted from the text since they are not relevant to the basis for the screening of this initiator.</p> <p>The opening paragraph of the Spurious HPI Actuation initiating event screening previously suggested that the reason the initiator was screened was based on the assumption of successful operator response to terminate the injection prior to occurrence of an overpressure trip. However, this is not the case. A detailed discussion of how this event was screened on a probabilistic basis (including operator response failure) is provided in the table. The misleading sentences in the opening paragraph were deleted.</p> <p>Steam line break is not screened on the basis of a recovery action, but rather on the following basis included in Table 3: "The probability of a steam leak occurring in this small section of piping near the ICW pumps, provides a qualitative basis for concluding that an initiator for this failure would not be significant."</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3919 (continued)				Spurious HPI Actuation screening involves operator action HHF1NOTT7P - Failure to Terminate Spurious HPI Prior to Reactor Trip, which is now quantified in the EPRI HRA Calculator including citations of procedures and timing assumptions that provide justification for the action.	
A1-3920	Resolved	IE-C6	<p>The screening performed in Table 3 of PRA-A1-01-001S06, Rev. 2 generally follows the conditions specified in SR IE-C6. However, the conditions in this SR are not explicitly involved, i.e., the 10e-8 frequency was used with an OOM argument to discount the initiator, not criteria (a).</p> <p>Possible Resolution: As the ASME/ANS PRA Standard is currently written, the screening criteria in IE-C6 needs to be used. The calculation for spurious HPI actuation needs to be checked. Add spurious HPI actuation due to spurious ESAS actuation as an additional initiating event.</p>	The only initiating event screening that relies on probabilistic analysis is for the Spurious HPI Actuation event. A detailed discussion of how this event was screened on a probabilistic basis (including the basis for meeting IE-C6 numerical screening criteria) is provided in Table 3 of PRA-A1-01-001S06.	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3923	Resolved	HR-C2	<p>The ANO-1 PRA Peer Review road map indicated that the pre-initiator events have been reviewed against plant-specific failures.</p> <p>Possible Resolution: To be assessed at CC II/III for this SR, a list of existing pre-initiator events at ANO-1 needs to be prepared, and it needs to be compared to the list in Table 2. Events not appearing in Table 2 would need to be added.</p>	<p>The data analysis update has included a review of Maintenance Rule Functional Failures and no new pre-initiator events have been identified.</p> <p>In addition, condition reports were reviewed for latent human errors. The attached list of condition reports reviewed is included in the attached link. None of the events would impact the HRA evaluations.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>
A1-3928	Resolved	SC-B5	<p>No documentation exists describing comparisons with similar plants or other plant specific codes to check the reasonableness and acceptability of the results of the thermal/hydraulic, structural, or other supporting engineering bases that support the success criteria.</p> <p>Possible Resolution: Perform a comparison with other plants and document.</p>	<p>A section has been added to the Success Criteria Notebook (PSA-ANO1-01-AS-03) to address the comparison of results.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3931	Resolved	IE-C14	<p>Analyst reviewed Calc. PRA-A1-01-001S08. There was no reference made to surveillance tests at power, if applicable, in which the ISLOCA pathway configuration would be changed from its routine configuration and alignment, e.g., 2 isolation valves instead of 3. Common cause mechanisms were discussed as not being applicable when they should have been included.</p> <p>Possible Resolution: Reference testing procedures and their frequency in order to more accurately account for the time when the ISLOCA pathway is in a different configuration (i.e., 2 valve isolation instead of 3). Consider common cause failure mechanisms, which also are related to 'state-of-knowledge' correlation (see QU-A3).</p>	<p>The ANO-1 Fire PRA model ISLOCA logic was revised to capture the potential for a human performance error in the restoration of the DHR isolation valve CV-1400/1401. LHF1LPITNA was added to the ISLOCA logic to address this issue.</p> <p>The value for this pre-initiator was calculated using the following formula.</p> $\text{Unavailability} = 4 * \text{TRT} / 365 * 24.$ <p>Where, TRT is time for the valve to be in open position per test.</p> <p>Per OP-1104.004 Supplement 1, the acceptance criteria for opening this valve is 11.8-15.9 seconds. Assuming that it takes approximately the same time to close the valve, an assumption of 2 minutes per surveillance is a reasonable estimate for TRT.</p> <p>Additional references and discussions added to the ISLOCA documentation (PSA-ANO1-01-AS-02) provided to address the CCF and surveillance testing.</p> <p>State of knowledge correlations are contained in sections 4.3 and Appendix A of the ISLOCA NB that is being developed for Rev. 5.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3932	Resolved	IE-C14	<p>The capability of secondary system piping appears to be such that once the isolation valves fail, the low pressure piping automatically fails.</p> <p>Possible Resolution: State that a conservative approach was taken by assuming automatic failure of secondary piping once it is exposed to high pressure, either via leak or rupture.</p>	Additional discussions have been added to the ISLOCA notebook (PSA-ANO1-01-AS-02) to address the conservatism.	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.
A1-3936	Resolved	SY-B4	<p>In reviewing the actual fault tree model, it was seen that the CCF terms were incorporated per the system modeling documentation (Supplement 11) and CCF documentation (Supplement 4). However, there was an inconsistency with what was stated in Supplement 4 and what was incorporated in the PRA model.</p> <p>Possible Resolution: Please resolve discrepancy between what was recommended in the common cause calculation and what was done in the PRA model.</p>	The CCF event probabilities were updated via PSA-ANO1-01-DA-02. These updated probabilities are used in quantification.	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3946	Resolved	QU-A3	<p>The state of knowledge correlation was not accounted for where it would make a significant difference, i.e., the ISLOCA analysis omitted common cause failure of check valves. Document PRA-A1-01-001S08 was reviewed to confirm this.</p> <p>Possible Resolution: Consider common cause failure mechanisms, which also may be related to 'state-of-knowledge' correlation (see QU-A3).</p>	<p>The impact of the state of knowledge correlation on the ISLOCA model was evaluated by developing a simple tree with two tops. The first top is an AND gate of the two check valve failures tied to the same type code. The second top is an AND gate of the two check valves not tied to the same type code. The tree is quantified and the point estimate mean of both tops is essentially the same (as expected). UNCERT is run for both tops using the same sample size and the same seed. The Uncertainty mean for the correlated check valves is significantly higher than for the non-correlated valves. This test shows that a correlation factor of 1.42E-6 is needed to account for the SOKC impact.</p> <p>Four events are added to the model to address this change:</p> <ul style="list-style-type: none"> • %FRDH14ASOK Shared Failure of DH-14A and DH-13A due to State of Knowledge Correlation 1.420E-06 • %FRDH14BSOK Shared Failure of DH-14B and DH-13B due to State of Knowledge Correlation 1.420E-06 	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3946 (continued)				<ul style="list-style-type: none"> • %FRDH17SOK Shared Failure of DH-14B and DH-17 due to State of Knowledge Correlation 1.420E-06 • %FRDH18SOK Shared Failure of DH-14A and DH-18 due to State of Knowledge Correlation 1.420E-06 <p>NUREG/CR-5102, Interfacing Systems LOCA: Pressurized Water Reactors, was used to determine if a common cause failure of check valve leakage or rupture is needed for the ISLOCA. Appendix B of this NUREG evaluates the failure probability of multiple failures and determined that a CCF event is not necessary.</p>	
A1-3948	Resolved	QU-F5	<p>A review of the Integration and Quantification Work Package (PRA-A1-01-001S02) and the FORTE Qualification Engineering Report (SA-01-001-01) did not reveal any documented software or quantification limitations that would impact applications.</p> <p>Possible Resolution: Document any known quantification limitations, and if none, state that there are no known limitations.</p>	<p>Section 3.0 of the Quantification notebook (PSA-ANO1-01-QU, ANO-1 Integration and Quantification Work Package) documents the Integration and Quantification Analysis Method. This section includes a table with the name of each software used, what function it was used for, and the documented qualification package for that software. (FTREX was quant engine used) Section 5.4 of the Quantification notebook describes limitations of the quantification methodology.</p>	<p>This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-3949	Resolved	QU-F6, LE-G6	<p>Attachment E of the Summary Report (PRA-A1-01-001R1) was found to list significant accident sequences and basic events. Attachment C lists the top 25 cutsets.</p> <p>Possible Resolution: Provide definitions for risk significant basic events, cutsets, and accident sequences within the Summary Report that comport with those listed in Section 1-2.2 of the ASME Standard.</p>	Section 6 of the Quantification notebook (PSA-ANO1-01-QU, ANO-1 Integration and Quantification Work Package) documents the definition of significant basic event (6.3), significant cutset (6.2) and significant accident sequence (6.1). The definitions and corresponding relevant model results fully satisfy the SR QU-F6 requirements.	This issue was resolved and therefore has no impact on STI change evaluations performed in accordance with the SFCP.
A1-5886	Unresolved	IFPP-B3, IFEV-B3, IFQU-B3, IFSN-B3, IFSO-B3, IFQU-A7	<p>No documentation of modeling uncertainties is provided in any of the internal flooding notebooks.</p> <p>Possible Resolution: Identify the applicable modeling uncertainties and document these uncertainties in the appropriate flooding notebooks to satisfy the requirements of this SR.</p> <p>For example, the flood area uncertainties could include the potential for leakage through penetrations that are not sealed as designed, undocumented flowpaths between areas, and equipment that is more susceptible to water intrusion than expected. These uncertainties could be discussed within PSA-ANO1-01-IF-WD and shown to be minimal based on confirmatory walkdowns.</p>	This is a documentation issue and additional discussion will be included in the IFA documentation.	This is a documentation issue and additional discussion of flooding sources of uncertainty is not expected to impact STI evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-5887	Unresolved	IFQU-A5	<p>The calculation of HEPs is documented in the quantification notebook (PSA-ANO1-01-IF-QU), attachment A. The HEPs are quantified using the EPRI HRA calculator and generally comply with the Capability Category II post-initiator HEP SRs noted in the HR-E, HR-F, and HR-G HLRs. However, the following discrepancies are noted:</p> <p>HR-E3, HR-E4, and HR-G5: Operator walkthroughs or talkthroughs of the flooding isolation actions were not performed.</p> <p>HR-G4: HFE timing assumptions for groups of scenarios is not consistently applied. Some utilize the most limiting time and others utilize average time available.</p> <p>HR-G6: A consistency check of the HEPs was not performed.</p> <p>HR-H2: Execution steps for several HFEs include execution recovery steps labeled "self-review". These do not appear to be actual procedural steps and there does not appear to be a valid cue that would initiate the recovery. 'self-review' is typically considered in the cognitive portion of an HFE only.</p>	<p>Additional discussion will be included in the IFA documentation and adjustments to some flooding HFE timing will be implemented as needed.</p>	<p>Additional discussion of flooding HEPs, or adjustments to some flooding HFE timing or execution steps is not expected to impact flooding results, but is expected to be assessed in case-by-case STI evaluations.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-5887 (continued)			Possible Resolution: Perform and document talkthroughs of the flooding mitigation actions. Clarify timing assumptions for HFEs and state the basis for whether the most limiting or average system window will be utilized in the analysis. Determine whether valid cues and procedural steps are present to justify execution recoveries and document the basis for those recoveries. Also perform and document a consistency review of the flooding HEPs.		
A1-5889	Unresolved	IFQU-A10	<p>LERF was not considered in the internal flooding PRA.</p> <p>Possible Resolution: Perform an evaluation of LERF due to flooding. When performing this evaluation, determine if any LERF sequences need to be modified.</p>	LERF considerations will be included in the IFA analysis and documentation.	The lack of a LERF model may impact decisions pertaining to STI changes. The Rev. 6 model update is expected to use the integrated model to quantify LERF including internal flooding. The impact of this finding is expected to be assessed in case-by-case STI evaluations.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-5890	Unresolved	IFQU-A8	<p>The internal events PRA model, updated by FRANX to include flooding impacts, properly considers both flooding impacts and random equipment failures and human errors.</p> <p>However, at least one instance was identified in which a modified internal events HEP was not properly included in the integrated model. See event EFW-HFC-FO-T41BX, which was supposed to be set to 1.0 in all auxiliary building flooding scenarios that occurred at elevation 386 feet and lower.</p> <p>Possible Resolution: Review the process used to incorporate the modified HEPs into the internal flooding model and re-quantify using the corrected HEP values.</p>	The impacted HEPs will be updated as needed in the IFA model and documentation.	<p>The incorrect HEP is expected to result in a very small change to the flooding results, as the majority of failures for all flood areas were captured. However, impact of this finding is expected to be assessed in case-by-case STI evaluations.</p>
A1-5891	Unresolved	IFQU-A8	<p>The internal events PRA model, updated by FRANX to include flooding impacts, properly considers both flooding impacts and random equipment failures and human errors.</p> <p>Attachment A of the quantification notebook identifies a flooding HEP to be included in certain flooding scenarios (IFL1-HFC-FO-CTM, to be included with initiating events %FLAB3350020CTM and %FLAB3350038CTM). However, this HEP is not included in the flooding PRA model.</p> <p>Possible Resolution: Review the process used to include the new flooding HEPs and ensure that all intended events were included.</p>	The impacted HEPs will be updated as needed in the IFA model and documentation.	<p>The incorrect HEP is expected to result in a very small change to the flooding results, as the majority of failures for all flood areas were captured. However, impact of this finding is expected to be assessed in case-by-case STI evaluations.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-5892	Unresolved	IFQU-A7	<p>QU-D is not met: There is no evidence that the sequence, cutset, importance, and consistency reviews, nonsignificant cutset review, or the comparisons to other plants were performed for the internal flooding analysis.</p> <p>Possible Resolution: Perform the specific analyses and evaluations indicated in the QU-D SRs.</p>	This is a documentation issue and additional discussion will be included in the IFA documentation.	This is a documentation issue and additional discussion of flooding results is not expected to impact STI evaluations performed in accordance with the SFCP.
A1-5893	Unresolved	IFQU-A7	<p>A parametric uncertainty analysis was not performed as required by QU-E1.</p> <p>Possible Resolution: Perform and document a parametric uncertainty analysis.</p>	This is a documentation issue and additional discussion will be included in the IFA documentation.	This is a documentation issue and additional discussion of flooding results is not expected to impact STI evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-5894	Unresolved	IFQU-A7	<p>It does not appear that HFE values were seeded to a higher level to ensure that cutsets with multiple HFEs are not truncated as required by QU-C1.</p> <p>Possible Resolution: Perform a systematic process to identify all multiple-HFE combinations that need to be considered for the HEP dependency analysis.</p>	The impacted HFEs will be updated as needed in the IFA model and documentation.	<p>The HEP seed values are expected to have minimal or no impact on the flooding results, as the risk-significant dependencies were captured. However, impact of this finding is expected to be assessed in case-by-case STI evaluations.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-5895	Unresolved	IFEV-A8, IFEV-A7	<p>Screening for scenarios was performed within PSA-ANO1-01-IF-AS for scenario groups that had very long operator response times (e.g. greater than 3 hours) due to the relative certainty that operators would be successful in isolating prior to equipment damage. No quantitative basis was provided in this case.</p> <p>Individual flood initiators were typically not screened out, except for maintenance related floods. In the case of maintenance related floods (see section 2.1.1 of PSA-ANO1-01-IF-IE), the screening criteria does not meet the numerical threshold of IE-C6.</p> <p>Maintenance event screening is currently documented as a $\sim 1\text{E-}05/\text{yr}$ frequency. Even with the proposed 0.1 factor to account for maintenance that breaches the pressure boundary, the criteria are still not met.</p> <p>Possible Resolution: Perform a more detailed review of maintenance practices and update documentation to show these events meet the screening criteria. Use scenario specific isolation failure data if necessary.</p>	This is a documentation issue and additional discussion of potential maintenance-related floods flooding mechanisms will be included in the IFA documentation.	Minimal or no impact on the results. This is a documentation issue only. This Unresolved F&O does not impact STI evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-5896	Unresolved	IFSN-A11, IFEV-A4	<p>The discussion of area 34 in PSA-ANO1-01-IF-AS does not appear to include the U2 SW return line in that area. This line needs to be included and the discussion should include the potential for a multi-unit event. The SW return header is a unit 2 pipe and will require action by U2 operators to realign SW to an alternate discharge path to stop the release and to prevent a loss of SW and therefore a unit trip on U2.</p> <p>Possible Resolution: Revisit potential initiating events in area 34 to determine if the U2 SW return line is applicable. Update accident sequence and initiating events as appropriate. Include operator response for isolation and realignment of SW discharge. Consider whether a U2 loss of SW event should be included in the U2 flood quantification.</p>	Potential initiating events, including multi-unit scenarios, will be updated as needed in the IFA model and documentation.	The impact of this finding is expected to be assessed in case-by-case STI evaluations.
A1-5897	Unresolved	IFQU-A7	<p>QU-B1: FRANX 4.2 Beta 2 was used for quantification. This software is not approved and final tested software. Since EPRI software testing was not completed, this software may contain bugs that would cause quantification results to be incorrect.</p> <p>Possible Resolution: Utilize approved "final" version of software for quantification or provide a review of the software version used that demonstrates that results are acceptable.</p>	This is a documentation issue and additional discussion will be included in the IFA documentation.	Additional software testing is not expected to impact the internal flooding results or STI evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A1-5898	Unresolved	IFQU-A5, IF-QU-A7	<p>Flood specific post-initiator HFEs were developed to support the internal flood quantification. These HFEs were documented in PSA-ANO1-01-IF-QU Attachment 1. The industry standard HRA calculator was used to perform the analysis.</p> <p>HR-G7: Dependency analysis for multiple HFEs appears to have been performed. However this analysis was not documented. Also, it is not clear whether the dependency analysis includes both IE and IF.</p> <p>Possible Resolution: Perform and document dependency analysis for multiple HFEs. Ensure the analysis includes combinations of internal events and internal flooding HFEs when applicable.</p>	This is a documentation issue and additional discussion will be included in the IFA documentation.	This is a documentation issue and is not expected to impact the internal flooding results or STI evaluations performed in accordance with the SFCP.

Table 3

List of SRs Assessed as CC-I in the ANO-1 Internal Events PRA Model

SR	Topic	Status	Importance to Application
LE-C1	Accident progression level of detail for containment challenges.	Updated to CC-II in model revision 5p0. Accident sequences consider contributors to large early release. Containment challenges for DCH and H2 burn are compared with plant specific containment fragility curves (See Tables 6.11-2 and 6.11-3 WCAP-16341-P).	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
LE-C2	Treatment of feasible operator actions after onset of core damage.	Updated to CC-II in model revision 5p0. Generic model extended to include SAMG mitigation and recovery actions. Actions limited to those that may impact LERF. Actions that impact LERF were determined to be depressurization of the RCS, re-establishment of off-site power and "bumping" the pump. ANO-1 guidance to allow bumping the pump is an important contributor to LERF and its basic event value is studied parametrically. Human action to depressurize the RV following core damage is considered. Recovery of offsite power is considered based on ANO-1 data for LOSP recovery. No additional core damage recovery factor is included.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
LE-C3	Credit for repair of equipment.	No maintenance or operator actions are credited in this evaluation for recovering initially failed High Pressure Injection (HPI) or any other equipment that would be capable of recovering the core within the reactor vessel.	Conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.
LE-C4	Realistic estimation of significant accident progression sequences resulting in large early release.	Updated to CC-II in model revision 5p0. Beneficial failures and outcomes considered for PSV fail to reseal. Evaluation based on steam cycles only; sensitivity studies provided. No fission product scrubbing considered.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

SR	Topic	Status	Importance to Application
LE-C5	System success criteria for significant accident progression sequences.	The success criteria for the systems are the same as Level 1. Additional success criteria are developed for Containment Isolation. Conservative bounding analyses used for DCH and hydrogen burn contributors. Conservative realistic methods are used for PI-TI-SGTR. SG tubes considered with an average flaw distribution. This likely overstates impact since few tubes have actually been plugged.	DCH and hydrogen burn contributors are not risk significant for most plants. Conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.
LE-C9	Credit for continued equipment operation or human actions under adverse environments.	The approach taken is WCAP-16341-P, which is an extension of NUREG/CR-6595. The NUREG considers operator action to depressurize without identifying a specific analysis to support PORV operation in the post core damage environment. The approach taken by ANO-1 does not address PORV operation in the post core damage environment but this is consistent with the NUREG.	Effects on operator actions and component availability in harsh environments are not expected to be significant for LERF for plants with large dry containments such as ANO-1. Therefore, conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.
LE-C10	Credit for continued equipment operation or human actions under adverse environments during significant accident progression sequences.	No requirements associated with CCI, credit is precluded by LE-C9.	See above.
LE-C11	Credit for continued equipment operation or human actions that could be impacted by containment failure.	Updated to CC-II in model revision 5p0. The model considers CHR containment failures as LATE unless the event is preceded by loss of recirculation cooling. In that event, SI is already inoperable and no credit is taken.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
LE-C12	Credit for continued equipment operation or human actions after containment failure during significant accident progression sequences.	Detailed review to reduce LERF contribution not performed.	Conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.

SR	Topic	Status	Importance to Application
LE-C13	Treatment of containment bypass and credit for scrubbing.	All core damage events involving a spontaneous SGTR, PI-SGTR, or a TI-SGTR event were conservatively assumed to lead to a large early release. No credit for scrubbing is taken for re-defining LERF events.	Conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.
LE-D1	Containment ultimate capacity.	The approach taken is WCAP-16341-P, which is an extension of NUREG/CR-6595. Used ANO-1 fragility curves from NUREG/CR-6475 for DCH and hydrogen burn assessments.	Conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.
LE-D2	Impact of containment seals, penetrations, hatches, and vent piping bellows as potential containment challenges.	The approach taken is WCAP-16341-P, which is an extension of NUREG/CR-6595. Generic assessment based on ILRT experience.	Conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.
LE-D3	Definition of containment failure location that affects classification as a large early release.	The approach taken is WCAP-16341-P, which is an extension of NUREG/CR-6595. Location effects not considered in determining LERF. All containment failures considered large.	Conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.
LE-D4	Interfacing system failure probability.	All core damage events involving an ISLOCA event are conservatively assumed to lead to a large early release.	Conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.
LE-D5	Secondary side isolation capability for accident progression sequences caused by SG tube failure resulting in a large early release.	The analysis has used a conservative evaluation (WCAP-16341-P) of secondary side isolation capability for significant accident progression sequences. For most sequences secondary isolation is assumed to fail. ANO-1 plant specific assessment performed by EPRI.	Conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.

SR	Topic	Status	Importance to Application
LE-D6	Analysis of thermally-induced SG tube rupture.	The approach taken is WCAP-16341-P, which is an extension of NUREG/CR-6595. Models based on NUREG-1570 and follow-on EPRI reports including reports with plant specific ANO-1 emphasis.	Conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.
LE-D7	Containment isolation analysis.	The RBI system is modeled realistically. Purge/venting strategies are not identified, actions are considered low probability.	Negligible LERF contribution expected from containment isolation failure. Therefore, conservative treatment is expected to have minimal impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.
LE-G3	Relative contribution of contributors to LERF (plant damage states, accident progression sequences, phenomena, containment challenges, containment failure modes).	Updated to CC-II in model revision 5p0. The significance of accident sequences and phenomena are documented.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

3.3 ANO-1 Fire PRA Model

3.3.1 Plant Changes Not Yet Incorporated

Similar to the internal events model, as part of the fire PRA evaluation for each STI change request, sensitivity cases are expected to be explored for areas of uncertainty associated with open items (peer review Findings for ASME/ANS PRA Standard CC II or plant changes) that would impact the results of the STI change evaluation, prior to presenting the results of the risk analysis to the IDP. A review of open MCRs against the ANO-1 fire PRA model showed that there are several ECs related to NFPA 805 modifications which have not yet been implemented. The internal events and internal flooding models are currently being updated with the NFPA 805 modifications. As stated above, each item is expected to be reviewed and assessed during the specific STI request.

3.3.2 Peer Review Facts and Observations

The ANO-1 fire PRA model has undergone several peer reviews, including a full scope and three focused-scope peer reviews. These reviews document the model quality and identify any areas with potential for improvement. The following assessments have been performed and documented for the ANO-1 fire PRA model:

- In October 2009, a Westinghouse Owners Group peer review was conducted under LTR-RAM-II-10-003 (Reference 11). There was a total of 54 F&Os, which included 41 findings, 13 suggestions, and no best practices. The conclusion of the review was that the ANO-1 Fire PRA methodologies being used were appropriate and sufficient to satisfy the ASME/ANS PRA Standard RA-Sa-2009. The review team also noted that the staff appeared to be applying methodologies correctly.
- In May 2012, a focused-scope peer review was conducted by the Kleinsorg Group to assess supporting requirements FSS-G3, FSS-G4, FSS-G5 and FSS-G6 of the ASME/ANS Combined PRA Standard. This focused-scope peer review is documented in Kleinsorg report 0021-0022-005 (Reference 12). The Kleinsorg Group provided a total of 8 F&Os, of which 6 were findings.
- In October 2012, a focused-scope peer review was conducted by Kazarians & Associates (Reference 13), which concentrated on the fire modeling for the SRs of FSS-A, C, D, E and H. The review was focused on the fire modeling parts of the Fire PRA. Only suggestions were provided from this peer review.
- In June 2014 a focused-scope peer review was conducted by Curtiss-Wright Sciencetech (Reference 14) which concentrated on fire HRA development and was performed against RG 1.200, Rev. 2. The peer review found that the ANO Unit 1 fire HRA was performed consistent with the guidance set forth in NUREG/CR-1921, Attachment B, "Detailed Quantification of Fire Human Failure Events Using the EPRI Fire HRA Methodology." There were 5 F&Os as a result of this review, 3 of which were findings.

3.3.3 Consistency with Applicable PRA Standards

As discussed in Section 3.1, the ANO-1 Fire PRA model was updated in 2016. Per Entergy procedures, all Entergy PRA models are required to meet current industry standards for PRA model development and documentation. Specifically, the Entergy PRA guidelines were developed to attempt to meet the ASME/ANS PRA standard (Reference 4) CC II of all SRs. Current Entergy PRA documentation includes an individual self-assessment that documents how each HLR and SR is met.

NUREG/CR-6850 guidance was the primary methodology used for the development of the fire PRA. The updated fire PRA in some cases used methodologies that extend beyond the guidance of NUREG/CR-6850. These methods, used in the ANO-1 Fire PRA Self Approval Model and discussed in Table 4, are considered extensions of the NUREG/CR-6850 methods and are documented via reference to approved NEI 04-02 frequently asked questions (FAQs) or other NUREGs. These references are:

- NUREG/CR-6850, Supplement 1, Rev. 0, "Fire Probabilistic Risk Assessment Methods Enhancements" (EPRI 1019259)
- NUREG/CR-7150, Vol 2, "Joint Assessment of Cable Damage and Quantification of Effects from Fire" (JACQUE-FIRE)
- NUREG-1921, Rev. 0, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines – Final Report"
- FAQ 14-0009, Rev. 1, "Treatment of Well-Sealed MCC Electrical Panels Greater than 440V"

The latest full scope peer review for ANO-1 fire PRA model was conducted in October 2009 using RG 1.200, Rev 2. In addition, focused scope peer reviews were performed in 2012 and 2014 on the FSS, fire modeling, and HRA elements of the fire PRA, resulting in a limited number of additional finding-level F&Os. Since then, a model revision was completed which ensured all the Finding-level F&Os from the peer review were addressed properly. Table 4 provides a listing of the finding-level F&Os related to the fire PRA and the acceptability of the finding-level F&Os in relation to this application. The information provided in Tables 4 and 5 summarizes the information provided in Attachment V of the ANO-1 NFPA 805 License Amendment Request (Reference 17) regarding disposition of the peer review Findings. The column labeled "Changes to Modeling Elements" has been added to provide information relating to changes to the ANO-1 fire PRA model from that provided in the subject letter. Table 5 lists SRs associated with the fire PRA. These SRs were assessed as CC-I only and the table provides the disposition of CC-I acceptability for this application.

As part of the fire PRA evaluation for each STI change request, sensitivity cases would be expected to be explored for areas of uncertainty associated with open items (peer review Findings for ASME/ANS PRA Standard CC II or plant changes) that would impact the results of the STI change evaluation, prior to presenting the results of the risk analysis to the IDP. At present, there are open items associated with the F&Os; however, the disposition of the F&O addresses the impacts to the fire PRA. Therefore, no sensitivity case is required.

Table 4

List of Finding F&Os against the ANO-1 Fire PRA Model

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	PP-A1	Fire PRA Peer Review Finding PP-A1-01: Based on the documentation, it is not clear that all inputs to the evaluation were considered in determining the failure probability. Input parameters include: tray/conduit, CPT/no CPT, cable type, and cable configuration. This finding is being assigned against A1 because the lack of documentation reveals that inputs were appropriately used in all cases.	This F&O is a duplicate to F&O CF-A1-02. Supporting Requirement PP-A1 is associated with defining the global analysis boundary and does not include failure probabilities. See the response to F&O CF-A1-02 for the actions to address this comment.	See changes as depicted for CF-A1-02.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
A1-5656	Resolved	PP-B2	<p>Fire PRA Peer Review Finding PP-B2-01: As discussed in Section 2.2 of Plant Partitioning and Fire Ignition Frequency, the method used to partition is based upon fire zones contained within the Fire Hazards Analysis. The barriers as described in the Fire Hazards Analysis are both rated and non-rated. Without reviewing each individual fire zone boundary within the Fire Hazard Analysis (FHA), there is no list of credited barriers that are not rated.</p> <p>ANO-1 needs to provide a list of barriers credited for fire compartment boundaries that are not rated and justify the credit for boundaries. This may be done through the evaluation of adequacy included in the multi-compartment analysis.</p>	<p>SR PP-B2 references NUREG/CR-6850, Chapter 1 for the acceptable criteria for justifying non-rated fire barriers. NUREG/CR-6850 discusses the use of fire compartments as “a well-defined enclosed room, not necessarily with fire barriers.” ANO references the FHA as a starting point for plant partitioning and all barriers (both rated and non-rated are defined in the FHA). The Plant Partitioning Task (CALC-08-E-0016-01, R0) assumes that fire protection features will be effective at containing a fire under most conditions. Fire protection features include fire-rated barriers, non-fire-rated barriers, active features such as water curtains, and in some cases spatial separation. The potential failure of a credited partitioning feature is addressed in the multicompartment analysis (MCA).</p> <p>The ANO FHA does not include any partitioning features, such as partial height walls, that are discussed in NUREG/CR-6850 as barriers that should not be credited. Nevertheless, this SR remains “open” as the current analysis only meets Capability Category I and would result in an identical finding upon re-review. The adequacy of the fire barriers is explicitly reviewed as part of the Multi-Compartment and Hot Gas Layer Analysis calculation (PRA-A1-05-009).</p>	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
A1-5657	Unresolved	PP-B3	<p>Fire PRA Peer Review Finding PP-B3-01: Spatial separation is identified as only credited for the Turbine Deck, Fire Zones 197-X and 2200-MM, (Report 0247-06-0006.03, Rev 1, Attachment A, Note 1). The note specifies that 1) no significant PRA components are located in the vicinity of the interface between these fire zones, and 2) the open turbine building and large associated volume will preclude any significant fire spread between these zones. Per NUREG/CR-6850, Volume 2, Pages 1-8, there are several considerations for the basis for using spatial separation as a boundary. The presence of PRA equipment, although important in later portions of the analysis, is not relevant with regard to spatial separation as a boundary. The open and large volume criterion is necessary, but not sufficient. Additional criteria that must be demonstrated include "minimal combustible fuel loads," and "free of ignition sources," among others. Therefore, there is insufficient justification documented to support this SR. In addition, two other fire compartments were identified (159-B and 2151-A) that credit spatial separation as a compartment boundary. The justification for use of spatial separation between these compartments was not explicitly identified in the reports. There is no apparent systematic review of PRA physical analysis units to identify when spatial separation is used or justified.</p> <p>ANO-1 needs to perform a system review of the PRA physical analysis unit to identify and justify when spatial separation.</p>	<p>This finding was not addressed in the partitioning effort; therefore, no change to the partitioning was performed as a result of this finding. The scenario development and MCA associated with these areas concluded that the spatial separation (lack of an actual barrier) did not impact fire results. No fires were judged to credibly breach the spatial separation and no hot gas layer potential exists.</p> <p>ANO has two areas (four total, two for each unit) that are separated into Unit 1 and Unit 2 fire zones that have no fire barrier between the units. The turbine deck is separated into Zones 197-X for Unit 1 and 2200-MM for Unit 2. Also, the Fuel Handling areas are separated into Zones 159-B for Unit 1 and 2151-A for Unit 2. The turbine deck area is very large and a hot gas layer would not develop due to a fire in this area. The MCA performed for ANO turbine deck fire zones screened using the NUREG/CR-6850 process.</p> <p>In the Fuel Handling Area, the area is relatively large and will not create a hot gas layer. The Fuel Handling areas are modeled as full room burn-ups in the scenario calculation and the MCA screens the fire spread to the other area.</p> <p>This SR remains "open" as the current analysis only meets Capability Category I and would result in an identical finding upon re-review. While the partitioning element only meets Capability Category I, the scenario and MCA document that this limitation has no impact on results.</p>	Original disposition remains valid.	While this issue remains unresolved, there is no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	PP-B5	<p>Fire PRA Peer Review Finding PP-B5-01: Documentation does not identify the active fire barriers that are credited in compartment separation. Through discussions, the intent of the evaluation is that where active barriers (e.g., fusible-link dampers) are rated as part of an overall rated wall, the barrier itself is justified as an adequate fire compartment boundary. This justification needs to be included in the documentation. Additionally, the identification and justification needs to be provided for active fire barrier components that are part of non-rated barriers. Evidence could not be found that a systematic method to identify/justify active fire barrier components was performed.</p> <p>One method to resolve this item is to provide justification for active fire barrier components that are included within an overall rated barrier and identify all active fire barrier components as part of non-rated barriers and provide justification why barriers are adequate (i.e., barrier configuration is considered during multi-compartment analysis).</p>	Active fire barriers such as fire dampers are credited in the FHA for fire zones and are subsequently included in the physical analysis unit (PAU) definitions. Failure of these active fire barriers is included in the MCA by assuming a failure probability of 0.0074 based on fire door failure. Non-rated barriers are addressed with a failure probability of 1.0 in the MCA.	<p>The disposition to this F&O no longer depicts the model of record. The information below better depicts the current model of record.</p> <p>During the RAI process, the model was revised to address the issue raised by this F&O.</p> <p>There are no active fire protection systems supporting the MCA fire barriers that require an actuation system that involves signals from cables or a detection system as part of any PAU boundary at ANO-1.</p>	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	PP-C3	<p>Fire PRA Peer Review Finding PP-C3-01: Documentation is not well developed for the identification and justification of non-rated barriers (see F&O PP-B2-01), spatial separation (see F&O PP-B3-01), and active fire barriers (see F&O PP-B5-01). Also, there is no documentation of the walkdown required in PP-B7 (see F&O PP-B7-01).</p> <p>ANO-1 should satisfy the resolution for F&Os PP-B2-01, PP-B3-01, and PP-B5-01, and document the process and the results.</p>	<p>The issues of non-rated barriers, spatial separation, and active barriers, are discussed above (PP-B2, PP-B3, PP-B5). Walk-downs of all non-NRC or Insurance commitment fire barriers have been performed to document the basis for credit taken for fire zone boundaries.</p> <p>Quantification of MCA probability has conservatively used the door failure probability from NUREG/CR-6850 Table 11-3 as the boundary failure mechanism for all zones without openings to adjacent fire zones. For zones with openings to adjacent zones, the boundary failure probability was set to 1.0 and the volume of the combined zones was used for assessing the time to HGL formation (PRA-A1-05-009).</p>	<p>The disposition to this F&O no longer depicts the model of record. The information below better depicts the current model of record.</p> <p>During the RAI process, the model was revised to address the issue raised by this F&O. Using fire protection drawings, if a doorway separated the adjacent zone from the exposing zone, the door failure probability was utilized along with type 2 (Fire Dampers) and Type 3 (Penetration Seals) summed into one probability.</p>	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		PP-C3 (continued)			If no doorway was located on the drawings, the analysis utilized the Type 2 damper barrier probability combined with Type 3 (Penetration Seals) barrier probability. If the drawings were unclear, a Type 1 (Door failure) probability was conservatively added in addition to Type 2 and Type 3.	
N/A	Resolved	PP-C3	<p>Fire PRA Peer Review Finding PP-C3-02: The fire compartments for ANO (Units 1 and 2) are listed in Table 2-2 of the Plant Partitioning and Fire Ignition Frequency development (ERIN Report 0247060006.01, Rev. 3, 10/2/09). For each fire compartment in the table, it would be useful to identify the unit number or if the compartment is a shared compartment (between the two units). Unit number or shared designation will facilitate Fire PRA applications, upgrades, and peer review. It is recognized that the fire zone numbering generally distinguishes between units, though not in all cases.</p> <p>ANO-1 should clearly identify the unit number or if the compartment is a shared compartment.</p>	<p>A note has been added to Table 2-2 of the Plant Partitioning and Fire Ignition Frequency calculation (CALC-08-E-0016-01, which is the Entergy calculation number for the ERIN Report referenced in the F&O) to indicate that the compartments without unit designators can be identified by the reference drawing number.</p> <p>Those with reference drawings starting with FP-1 are Unit 1 compartments while those with reference drawings starting with FP-2 are Unit 2 compartments.</p>	No change. Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	ES-C1	<p>Fire PRA Peer Review Finding ES-C1-01: The HRA Notebook (Report 0247-06-0006.03-U1, Rev. 0) considers instrumentation in terms of providing cues for Operator actions, and determining feasibility. The instrumentation credited is identified in Appendix A associated with the HRA Basic Event and the related cue. Several different methods for providing operator cues are listed (i.e., different instrumentation sets). Although it is noted if some of the options are not Appendix R instrumentation, it should be clearly indicated which option is the credited instrumentation. See F&O ES-D1-01 and ES-D1-02. Therefore, SR ES-C1 is judged as not met.</p> <p>ANO-1 needs to identify instrumentation relevant to operator actions for HFEs to account for the context of fire scenarios in the Fire PRA to meet SR ES-C1. ANO-1 needs to clearly indicate which option is the credited instrumentation associated with HRA Basic Events.</p>	<p>The Fire PRA HRA Notebook (PRA-A1-05-007, which is the Entergy report referenced in the F&O), Attachment A, provides multiple operator cues for performing the operator actions. These cues show that the operator actions have sufficient diversity so that failure of a single instrument or instrument train will not prevent the operators from performing the action. Attachment B provides a simulator review of the indicators on the control panels and the reliance on instruments. In addition, the major operator actions are driven by the EOPs. EOPs do not list specific instruments for performing the actions.</p>	Original disposition remains valid.	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	ES-C2	Fire PRA Peer Review Finding ES-C2-01: There is no evidence that a systematic review of indications that could result in an undesirable operator action were identified and dispositioned. A review of control room instrumentation should be performed to identify possible areas where spurious indications of a single instrument could mislead the operator into performing undesirable actions is needed to meet CC-II.	Attachment B of the ANO-1 FPRA HRA Notebook (PRA-A1-05-007) is "Fire PRA Simulator Review – ANO 1." One of the specific items addressed during the review was to: "Identify critical indicators where fire damage to sensing devices, cables or other loop components could result in misleading information that may cause significant confusion to the operators and thereby degrade their effectiveness in the performance of tasks that are required."	The disposition to this F&O no longer depicts the model of record. The information below provides the correct information relating to the reference in the disposition. Revised Calc – PRA-A1-05-015 contains Appendix B, "OPERATOR INTERVIEWS AND SIMULATOR EXERCISES" with the relevant information.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
N/A	Resolved	CS-B1	Fire PRA Peer Review Finding CS-B1-01: A review of fires that could result in the loss of coordination through the loss of control power (through fire damage to breaker control cables) was performed. No specific analysis for this was performed. However, a review of the circuit design indicates that this is unlikely to exist based upon circuit design (multiple fuses) and cable routing. A specific review of this condition should be performed to confirm control and control power cables do not preclude operation of credited equipment.	Section 4.4 of the Component and Cable Selection Report (PRA-A1-05-003), provides documentation that all circuits and electrical distribution buses credited in the fire PRA have been analyzed for proper over-current coordination and protection. A description of the processes used is included via reference to Upper Level Document ULD-0-TOP-12, "ANO Unit 1 and 2 Electrical Protection/Coordination," Rev. 3.	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	PRM-A3	<p>Fire PRA Peer Review Finding PRM-A3-01: The one-top fire PRA model for ANO-1 was not complete and thus not available for quantification or review. Determining dominant contributors and sequence frequencies beyond the scenario level is problematic without a one-top fire model. Create the one top fire model and benchmark results against the FRANC results for use in quantifications and review.</p>	<p>The ANO-1 fault tree model used for FPRA application has been completed. Section 14 of the Fire Scenarios Report (PRA-A1-05-004) provides a detailed description of the ANO-1 databases utilized to generate, document and quantify the fire PRA model. This section includes reference to the specific fault tree model used.</p>	Original disposition remains valid.	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>
N/A	Resolved	PRM-B2	<p>Fire PRA Peer Review Finding PRM-B2-01: The PRA Peer Review for the ANO-1 internal events PRA was performed the first week of August 2009. As such, there has been insufficient time to reconcile the F&O that could have an impact on the Fire PRA. However, the ANO PRA team has reviewed all of the F&Os from the ANO-1 internal events PRA peer review. They have identified five F&Os that have the potential to impact the fire PRA, and developed action plans for their disposition. The F&Os from the internal events ANO PRA peer review that could impact the Fire PRA have not yet been implemented. ANO-1 needs to implement the action plan that has been developed to reconcile the ANO-1 internal events PRA F&Os that could impact the fire PRA.</p>	<p>The ANO-1 Internal Events Peer Review was performed in August 2009. The ANO-1 Fire PRA Peer Review was performed in late October 2009. Based on the limited time between the peer reviews, ANO did not have time to incorporate internal events F&Os before the Fire PRA review. As stated in this F&O, a limited number of internal events findings were determined to impact the Fire PRA. These F&Os were subsequently incorporated in the FPRA.</p> <p>Table 2 provides details about the internal events F&Os. The details provided in Table 2 include the status of each F&O and each finding's potential impact.</p>	Original disposition remains valid.	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	PRM-B5	<p>Fire PRA Peer Review Finding PRM-B5-01: There is no evidence of a review the accident sequence models to determine whether sequences need to be added or changed. This SR requires a REVIEW of the corresponding accident sequences and there is no objective evidence that this review was performed, although there does not appear to be any modified accident sequences and the staff confirmed that no sequences were modified.</p> <p>ANO-1 needs to document the review of the corresponding accident sequences for addition or modification.</p>	<p>Section 4.5 of the ANO-1 FPRA Component and Cable Selection Report (PRA-A1-05-003) discusses the Plant Response Model, including a dedicated section discussing success criteria. Appendix D discusses various accident sequence types and how they would or would not apply for the FPRA. Additional comments on the internal events model are discussed in Appendix F. The details provided in the component and cable selection document satisfy the PRM-B5 Supporting Requirement.</p>	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FSS-C1	<p>Fire PRA Peer Review Finding FSS-C1-01: Assignment of characteristics to fire scenarios does not meet CC II. In order to meet CC II, a two-point fire intensity model must be used to assign characteristics to the ignition source. Furthermore, per Note 2 of Table 4-2.6-4 (c) of the ASME/ANS RA-Sa-2009 Standard, CC II requires, as a minimum that the determination of minimum fire intensity capable of causing fire spread and/or damage to at least one member of the target set. Then the two-point fire intensity model is applied to characterize the damaging fires (i.e., fires above the minimum damage intensity). Therefore, this SR is judged to be met at CC I.</p> <p>A two-point fire intensity model must be used to assign characteristics to the ignition source to meet CC II.</p>	<p>Traditional multi-point heat release rate treatment was not applied in the ANO-1 analysis. Rather than the repetitive analysis inherent in a multi-point heat release rate treatment, the Conditional Probability of Propagating Fire factors specified for vented panels provide a multi-point treatment for vented panels based on a split fraction developed from the EPRI Fire Events Database. This split fraction specifies the fraction of fires impacting only the ignition source panel versus those fires which impact targets within the zone of influence of the panels. This approach provides a definitive means of differentiating between significant and limited fires that will not be significantly impacted by potential future refinements in ignition frequency and heat release rate.</p> <p>Section 16 of the Fire Scenarios Report (PRA-A1-05-004) discusses the use of generic fire modeling versus detailed fire modeling and justifies the ANO-1 approach for the FPRA application.</p>	<p>The disposition to this F&O no longer depicts the model of record. During the RAI process, the model was revised to address the issue raised by this F&O.</p> <p>The following information was obtained from the ANO-1 SE (ML16223A481). In PRA RAI 01.c (Reference 21), the NRC staff requested that the licensee explain its approach and whether the split fraction referred to in the resolution to FSS-C1-01 implied that the "Panel Factors method" (Reference 105), not accepted by NRC was used.</p>	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		FSS-C1 (continued)			<p>In its response to PRA RAI 01.c (Reference 10), the licensee clarified that the "Panel Factors" method was not used but rather generic fire modeling treatments (GFMTs) were used to define a three-point fire model to meet SR FSS-C1. The licensee explained that the first fire in the three-point fire treatment is a non-severe fire in which the source panel and the cables terminating at the source panel are damaged but the nearest target is not damaged. The second fire in the three-point fire treatment is a severe fire in which all targets within the 98th percentile zone of influence (ZOI) are impacted.</p>	

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		FSS-C1 (continued)			<p>The third fire of the three-point fire treatment is a fire that results in a hot gas layer (HGL) exceeding a 80 degree Centigrade (°C) criterion in which all targets in the fire zone are conservatively assumed to be damaged. These three fire models are discussed in the licensee's response to FM RAI 01.f discussed in SE Section 3.4.2.3. The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that it used a multiple-point fire intensity and duration model encompassing low likelihood but risk-contributing fire events consistent with the requirements in the PRA standard.</p>	

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FSS-C2	<p>Fire PRA Peer Review Finding FSS-C2-01: The generalized models used to support most of the significant fire scenario evaluations use peak heat release rates. For example, 8 of the top 10 fire scenarios listed in the CDF quantification results presented in Appendix A of the Summary Report (Entergy Report 0247060006.06-U1, Rev. 0, 9/11/09) are for "Base Scenarios," which are the equivalent of a fire safe shutdown analysis exposure fire in which everything in the compartment is assumed to fail at the compartment frequency without considering time-dependent fire growth. Time-dependent fire growth is considered on a limited basis, such as for the main control room abandonment scenario and for ventilated cabinets that are located in zones equipped with automatic detection, where credit may be taken for suppression by the fire brigade prior to sustaining external target damage (Fire Scenario report, Entergy Report 0247-06-0006.05-U1, Rev. 0, 9/11/09). Therefore this SR is met at CC I.</p> <p>Time-dependent fire growth should be considered for more of the significant fire scenarios, which are mostly evaluated using peak heat release rates to meet CC II. Expand use of time-dependent fire growth to additional significant fire scenarios as appropriate.</p>	<p>Since the ANO-1 FPRA Peer Review, ANO has made several refinements within the reviewed methodology to remove some conservatism and reduce overall CDF and LERF. These refinements include:</p> <ul style="list-style-type: none"> • developing more detailed fire scenarios • refining the components failed within the scenario • refining the fire HRA events and JHEPs. <p>The use of fire growth curves are not part of the Generic Fire Modeling Treatments used for ANO-1. Section 16 of the Fire Scenarios Report (PRA-A1-05-004 which is Entergy calculation for ERIN Report 0247-06-0006.05-U1) discusses the use of generic fire modeling versus detailed fire modeling, and justifies the ANO approach for the FPRA application (and only meeting Capability Category I).</p>	<p>The disposition to this F&O no longer depicts the model of record. See revised disposition for F&O FSS-C1-01.</p>	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FSS-C7	<p>Fire PRA Peer Review Finding FSS-C7-01: A multiple suppression path is modeled for the cable spreading room. Proper modeling appears to have been performed for the cable spreading room in the self-assessment for FSS-C7. This information should be formally documented with appropriate references. Documentation of calculation needs to be included in Fire PRA documentation.</p> <p>Document the calculation and include appropriate references.</p>	<p>Section 9.0 of the Fire Scenario Report (PRA-A1-05-004) documents the "Credit for Suppression and Detection Systems" and explicitly outlines how the NSP is calculated for the Cable Spreading Room (including appropriate references). Explicit credit for suppression and detection systems is taken for the Cable Spreading Room fire scenario only.</p>	<p>The disposition of this F&O is not accurate in relation to the current model of record.</p> <p>Multiple zones credit suppression and detection. Section 9.0 of the Fire Scenario Report (PRA-A1-05-004) documents the "Credit for Suppression and Detection Systems". Additionally, the MCA/HGL analysis (PRA-A1-05-009) provides information relating to the determination of suppression and detection when credited.</p>	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FSS-D1	<p>Fire PRA Peer Review Finding FSS-D1-01: Simplified fire modeling is performed as described in Attachment B of the ANO - Unit 1 Fire Scenario Report (Report 0247-06-0006.05-U1 Revision 0). There has been no further area-specific modeling done with more sophisticated tools to determine if significant risk contributors could be reduced or if the conservative values are bounding for all dominant scenarios. The potential may exist for nonconservative scenarios as well as risk reductions in the significant scenarios.</p> <p>Investigate further into whether or not use of more sophisticated modeling tools would change the results for the dominant fire scenarios in the higher risk areas, such as Fire Zone 99-M.</p>	Section 16 of PRA-A1-05-004, which is the referenced report in the F&O, explains and justifies the ANO use of the Generic Fire Modeling approach instead of a more detailed approach. This approach is based the Zone of Influence (ZOI) dimensions for each heat release rate bin on the value that produced the largest distance. In the absence of specific data, this is a conservative approach.	The disposition to this F&O no longer depicts the model of record. See revised disposition for F&O FSS-C1-01	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FSS-D3	<p>Fire PRA Peer Review Finding FSS-D3-01: Fire growth and propagation within cable trays are not explicitly treated in the HGL calculations. Page 34 of ANO-1 Fire PRA Summary Report says: "This approach incorporates conservatism in the time to HGL impact that, along with the conservatism of the heat release rates used, will envelope impacts due to additional heat release rates introduced by the ignition of cable trays external to the initial ignition source." Fire growth and propagation within the cable trays can add energy to the fire that would affect the HGL calculations. Although conservatism in the Heat Release Rate (HRR) may envelope this additional added heat, this is only assumed.</p> <p>Quantitatively evaluate fire growth and propagation within cable trays for the HGL calculations.</p>	Section 16 of the Fire Scenarios Report (PRA-A1-05-004) discusses the use of generic fire modeling versus detailed fire modeling and justifies the ANO approach for the FPRA application. This approach ensures ANO meets only the Capability Category I for FSS-D3.	<p>The disposition to this F&O no longer depicts the model of record. See revised disposition for F&O FSS-C1-01. Additionally, from the ANO-1 SE (ML16223A481)</p> <p>In its response to FM RAI 01.c(i) (Reference 11), the licensee stated that it used the FLASH-CAT model to calculate the HRR increase due to fire propagation in cable trays, and that it determined the expanded vertical and horizontal ZOI based on the total HRR of the ignition source and secondary combustibles using the methods described in the GMFTs.</p>	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		FSS-D3 (continued)			The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee properly accounted for the HRR contribution of secondary combustibles (cable trays) in determining the ZOI.	
N/A	Resolved	FSS-D7	<p>Fire PRA Peer Review Finding FSS-D7-01: The assessment of unavailability is, in part, based on fire protection program controls for implementing compensatory actions for out of service systems. The types of compensatory measures for out of service detection and suppression systems are given in the TRM. For detection systems (Section 3.3.6 of the TRM), hourly roving fire watches are used when less than 50% of the detectors in a zone are operable. For sprinkler systems (Section 3.7.9 of the TRM), an hourly fire watch is established if detection is operable in the area, otherwise, a continuous fire watch is established. Per NUREG/CR-6850, Appendix P, only continuous fire watches can be used for crediting availability of detection and suppression systems. Consider removing crediting of hourly fire watches and include component-specific unavailability data.</p>	<p>Section 9.0 of the ANO-1 Fire Scenario Report (PRA-A1-05-004) details credit for detection and suppression systems. Explicit credit for suppression and detection systems is taken for the Cable Spreading Room (Fire Zone 97-R) fire scenario only. Per Technical Requirement for Operation (TRO) 3.3.6.B, the failure of a detector in this zone would require the automatic suppression system in this area to be declared inoperable per TRO 3.7.9.A, which requires a continuous fire watch to be established within 1 hour. A review of plant maintenance history shows that limited unplanned maintenance has been performed on this detector in the past 20 years. Therefore, the unavailability of this system is very low and is considered to be enveloped by the system unreliability data taken from NUREG/CR-6850.</p>	<p>Due to revisions in the model, the disposition to this F&O no longer depicts the model of record. The following response better depicts the current Fire PRA model.</p> <p>Section 9.0 of the ANO-1 Fire Scenario Report (PRA-A1-05-004) details credit for detection and suppression systems. There are several fire zones in which detection is credited.</p>	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		FSS-D7 (continued)		There is no credit taken for hourly fire watch.	Per Technical Requirement for Operation (TRO) 3.3.6.B, the failure of a detector in this zone would require the automatic suppression system in this area to be declared inoperable per TRO 3.7.9.A, which requires a continuous fire watch to be established within 1 hour. There is no credit taken for hourly fire watches in these zones.	

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N/A	Resolved	FSS-G2	<p>Fire PRA Peer Review Finding FSS-G2-01: Embedded in the analysis is the assumption that a fire is always controlled within 20 minutes such that a hot gas layer (HGL) will not form beyond 20 minutes. This assumption is based on the concept that the fire brigade would arrive within 20 minutes and successfully mitigate the effects of the fire within that time by opening doors or suppressing the fire. This is treated in the model as a 1.0 probability. Issues associated with this assumption include: 1) no evidence is provided that the HGL temperature would not continue to increase following the opening of a door, and 2) this evaluation does not consider the probability distribution of brigade response time coupled with the actions that would be taken such as what is considered in NUREG/CR-6850 and Frequently Asked Question (FAQ) 50. Non-conservative screening methodology that may screen significant compartments.</p> <p>One possible resolution would be to credit a distributed manual suppression probability based upon actual time for hot gas layer development.</p>	The updated Multi-Compartment/Hot Gas Layer Analysis (PRA-A1-05-009) uses a distributed manual suppression probability based on 20-, 30-, 60-minute HGL growth rates. The updated MCA/HGL report develops non-suppression probabilities based on FAQ 08-0050 (FAQ 50) guidance.	<p>The original F&O disposition does not reflect the current model of record. The following response better depicts the current Fire PRA model.</p> <p>The manual non-suppression probability (NSPms) is calculated using a convolution process which applies the non-suppression curve to each bin of the applicable NUREG/CR-6850, Appendix E (Tables E-2 through E-9) heat release rates (HRR), with the time to hot gas layer varying based on the heat release rate associated with each bin. This NSPms is multiplied by the fraction of the total probability distribution (severity factor) applicable to each discrete bin of the HRR distribution and summed.</p>	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		FSS-G2 (continued)			<p>This final value gives a manual non-suppression probability (NSPms) for each evaluated ignition source. The time to HGL is determined for each heat release rate bin using a curve fit of the available time to hot gas layer data for the various configuration heat release rate (single cable bundle panel, multiple cable bundle panel, motor).</p> <p>Additionally, the 20-30-60 min timing intervals have been replaced with detailed timing analysis for each fire scenario event tree.</p>	

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N/A	Resolved	FSS-G2	<p>Fire PRA Peer Review Finding FSS-G2-02: The multi-compartment analysis assumes that for each fire scenario, the only source of heat contributing to the hot gas layer is the heat from the cabinet (based upon a 98% HRR). This does not account for additional heat due to potential fire spread to other combustibles including cable. It is noted that a conservative (98%) HRR rate is used for the cabinet. However, it is not demonstrated that this is bounding (i.e., 1.0 probability) when considering the HRR over time due to fire spread. This could result in non-conservative screening of compartments – this may be compounded by the issue identified by F&O FSS-G2-01. One method to resolve this is to consider the HRR based on generic bounding or compartment specific configuration of cabinets vs. cables and described in NUREG/CR-6850 and FAQ 49.</p>	<p>The updated Multi-Compartment/Hot Gas Layer Analysis (PRA-A1-05-009) uses more detailed methods for analyzing hot gas layer development and growth. The updated methods account for both fire spreading and additional combustibles.</p> <p>The listed details are in Attachment A of the Multi-Compartment and Hot Gas Layer Analysis. The supplemental information relates to additional hot gas layer tables generated for transient fires, specific steady and peak heat release rate values, and scenarios that involve secondary combustibles.</p>	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
N/A	Resolved	FSS-G2	<p>Fire PRA Peer Review Finding FSS-G2-03: The analysis assumes (with the exception of the control room) that all barriers have no openings. Cases were identified where other openings in fire compartment barriers exist, e.g., 73W. A systematic effort is needed to identify these openings so that they can be accounted for in the multi-compartment analysis. This is a follow-on to the issue identified in F&O PPB3-01. The review indicates that it is likely on few compartments have such openings. Identify these compartments as part of Plant Boundary and Partitioning. Include the openings and impact of openings in the Multi-compartment analysis to justify low significance.</p>	<p>The updated Multi-Compartment and Hot Gas Layer Analysis (PRA-A1-05-009) methodology addresses openings between fire compartments.</p> <p>Openings between fire compartments are identified and their impact evaluated for every fixed ignition source analyzed in the Multi-Compartment Analysis.</p>	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

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N/A	Resolved	FSS-G2	<p>Fire PRA Peer Review Finding FSS-G2-04: The hot gas layer analysis is based on an actual fire compartment volume (area of room times height) where the fire modeling used to account for the amount of heat necessary to cause a hot gas layer is based upon the available hot gas layer volume. To evaluate the hot gas layer, it was assumed that the fire was at the floor level. This may result in non-conservative estimates for the amount of heat necessary to result in a hot gas layer. This methodology results in non-conservative heat requirements to cause a hot gas layer.</p> <p>Adjust the assumed room volumes in the screening process based upon the available hot gas layer volume as opposed to the full room volume.</p>	<p>The Multi-Compartment and Hot Gas Layer Analysis has updated the methodology in the following manner. The new approach assumes the ignition source/fire has a base 8 ft off the floor. The new methods also calculate room volume above the source for HGL impact (Attachment A of PRA-A1-05-009).</p>	<p>The disposition to this F&O does not reflect the current model of record. The information below is provided from the MCA/HGL report.</p> <p>Electrical Panel Fires are assumed to be seven feet high with fires at the top of the panel, reducing the total height of the room associated with the volume of air above the panel available for hot gas layer heat up by seven feet.</p>	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>
N/A	Resolved	FSS-H4	<p>Fire PRA Peer Review Finding FSS-H4-01: There is no apparent documentation that the technical bases for input values used in the fire modeling were validated by plant walkdowns or other methods. This SR requires documentation of a technical basis to be established for fire modeling tool input values given the context of the fire scenarios being analyzed. This was reported to be performed as part of the walkdowns for scenario development, but not documented.</p> <p>Document that the technical basis for fire modeling input values were validated in the context of the fire scenarios analyzed.</p>	<p>Attachment D of the Fire Scenarios Report (PRA-A1-05-004) is a Walkdown Workbook. This attachment provides the basis for FRANC inputs. Attachment A-2 of the Scenarios Report is the Scenario Development Walkdown Summary. These two attachments (with some additional information in other Scenario Report attachments) contain the details to support that all scenario development inputs were validated by walkdowns.</p>	<p>Original disposition remains valid.</p>	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	IGN-A5	<p>Fire PRA Peer Review Finding IGN-A5-01: Table 3-2: some IEFs are not calculated on a "per reactor-year" basis. All NUREG/CR-6850 ignition frequencies should be updated with reactor critical years in order to obtain the correct ignition frequencies per this SR. IGN-A5 requires generic fire ignition frequencies or plant-specific fire frequency updates on a reactor-year basis. This is not done for the following ignition frequency bins: 1, 4, 8-10, 12-16c, 17-19, 21, 23, 26, and 30.</p> <p>Update the "all-mode" Ignition Frequencies from NUREG/CR-6850 with critical years as opposed to calendar years.</p>	<p>Fire PRA Plant Partitioning and Fire Ignition Frequency Development calculation (CALC-08-E-0016-01 Table 3-2) has been changed to show that all bins were updated on a reactor-year basis.</p>	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
N/A	Resolved	IGN-B5	<p>Fire PRA Peer Review Finding IGN-B5-01: Section 1.1 of Report 0247060006.01 Revision 3 contains only one assumption. However, a text search of the document resulted in a number of additional instances of assumptions buried in the text. One was an assumption that the ignition frequencies were log-normally distributed, one was the assumption that the compartments were assigned in accordance with the generic sources, one was the assumption that junction boxes were uniformly distributed, and a general assumption for a number of the events that they occurred at power.</p> <p>All assumptions pertaining to the ignition frequency calculation should be explicitly captured in Section 1.1, with the possible exception of the "at-power" assumption for individual events. The assumptions should also be reviewed for completeness.</p>	<p>The ANO Fire Probabilistic Risk Assessment - Plant Partitioning and Fire Ignition Frequency Development (CALC-08-E-0016-01) contains an updated section on assumptions.</p>	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

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N/A	Resolved	CF-A1	<p>Fire PRA Peer Review Finding CF-A1-01: The tables in NUREG/CR-6850 were used to determine cable failure probabilities. The most conservative value from this table was chosen. The analysis does not account for the potential for both inter and intra cable short probabilities. For example, item 32-K, LMV101414K was assigned a failure probability of 0.3 for intra cable hot short. This value does not include the potential for inter cable hot short of .03 (total probability of 0.33). Since the highest failure probability in the table is used - this is typically the intra cable failure probability, the impact of excluding the inter cable failure probability is relatively small.</p> <p>Review cables where failure probabilities other than 1.0 are credited and ensure the appropriate inter and intra cable short probabilities are applied.</p>	<p>Section 12.0 of the Fire Scenario Report (PRA-A1-05-004) discusses cable failure probabilities. The section outlines the application of inter and intra cable short probabilities.</p> <p>For a circuit with a CPT, a bounding hot short probability of 0.33 is used which includes both intra- and inter-cable hot shorts. For a non-CPT circuit, a bounding hot short probability of 0.66 is used which includes both intra- and inter-cable hot shorts.</p>	<p>The disposition to this F&O does not reflect the current model of record. The information below depicts the current modeling and use of a new methodology.</p> <p>The NUREG/CR-6850 data associated with Hot Shorts has been revised to reflect the probabilities associated with NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)". The use of NUREG/CR-7150 will not affect the original disposition of this F&O.</p>	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

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N/A	Resolved	CF-A1	<p>Fire PRA Peer Review Finding CF-A1-02: Based on the documentation, it is not clear that all inputs to the evaluation were considered in determining the failure probability. Input parameters include: tray/conduit, CPT/no CPT, cable type, and cable configuration. This finding is being assigned against A1 because the lack of documentation reveals that inputs were appropriately used in all cases.</p> <p>Document the specific configuration inputs used in justifying the chosen failure probabilities from NUREG/CR-6850 failure table probability and validate the chosen probabilities.</p>	<p>The NUREG/CR-6850 tables used for hot shorts are Tables 10-1 and 10-2, which are associated with thermoset cables. The remaining tables are associated with thermoplastic cables (Tables 10-3 and 10-4) or armored cables (Table 10-5). Calculation CALC-ANOC-FP-09-00019 identified only 10 thermoplastic cables for ANO with only two of these cables associated with ANO-1.</p> <p>Details about parameters pertaining to cable & circuit failure probabilities are documented in the Fire Scenario Report (PRA-A1-05-004). The cable at ANO is type IEEE-383 and the damage threshold for this type is specified in NUREG/CR-6850 (Section 6 of the Scenario Report). Section 12.0 of the Fire Scenario Report addresses the failure data applied for CPT/non-CPT circuits.</p>	<p>The NUREG/CR-6850 data associated with Hot Shorts has been revised to reflect the probabilities associated with NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)". The use of NUREG/CR-7150 will not affect the original disposition of this F&O.</p>	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

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N/A	Resolved	HRA-A2	<p>Fire PRA Peer Review Finding HRA-A2-01: To address an excess spray MSO, ANO-1 an HFE, RHF1RCPSXP, was added directly to the model logic. In this logic, RHF1RCPSXP feeds into an AND gate and has a value of "0.0" which in effects kills the entire logic structure. This HFE shows up in the TAGBE file tagged as "N2" which means it is ignored. It does not show up anywhere in the HRA report, the ExcludedEvents file or the AlteredEvents table so there is no definition or characterization that is traceable.</p> <p>Undocumented HFE that appears to be impacting logic. If RHF1RCPSXP is not used, rather than setting its value to 0.0, remove it from the model. If it is a valid HFE, it needs to be identified and fully characterized in the HRA report and the correct value needs to be calculated.</p>	<p>The modeling error has been corrected. The new event (RHF1RCPSXP) has a default value of 1.0. It is only changed if changed in the altered events table (in FRANC) if it is relevant to a specific case. With the default value of 1.0, it is does not disrupt quantification of the other logic in the AND gate for other cases.</p>	Original disposition remains valid.	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>
N/A	Resolved	HRA-A3	<p>Fire PRA Peer Review Finding HRA-A3-01: There is limited direct evidence that a systematic review of fire scenarios was performed to identify undesirable operator action that could result from spurious indications (See F&O ES-C2-01). The evidence in Attachment E of the HRA report is implicit. A review of each fire scenarios is needed to identify undesirable operator action that could result from spurious indications of a single instrument for CC II.</p> <p>Perform a systematic review of fire scenarios to identify undesirable operator action that could result from spurious indications of a single instrument, per SR ES-C2.</p>	<p>Attachment C of the ANO-1 FPRA Human Failure Events Notebook (PRA-A1-05-008) contains the systematic review of fire PRA credited operator actions. The Attachment contains the results of interviews with experienced ANO-1 operations personnel.</p> <p>Experienced operators were asked a series of questions about each HEP credited in the model. The questions included – description of the action, which procedures apply, what instruments/signals are available, time available and/or required to take action, location (inside or outside control room), and if there are any special considerations.</p>	Original disposition remains valid.	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		HRA-A3 (continued)		The FPRA HRA Notebook (PRA-A1-05-007), Attachment A, provides multiple operator cues for performing the operator actions. These cues show that the operator actions have sufficient diversity so that failure of a single instrument or instrument train will not prevent the operators from performing the action and reduces the likelihood of inadvertent/undesirable actions.		
N/A	Resolved	HRA-B3	Fire PRA Peer Review Finding HRA-B3-01: The new human failure events (HFEs) as identified in the FRANC "ALTEREDEVENTS" table are not developed as per HRA-B3 (nor identified in HRA-B2). These HFEs need to be "processed" to (1) determine the viability of the HFE – can it be performed, are there cues, etc., and (2) satisfied the HR supporting requirements (SRs) from Section 2 of the ASME/ANS PRA Standard (internal events). Note that the HFEs identified in HRA-B1 (from the internal events PRA) would have been assessed at CC III. The table in Appendix A of ANO-1 Fire Probabilistic Risk Assessment Human Reliability Analysis (HRA) Notebook (Report 0247060006.03-U1, Revision 0), September 2009 deals with timing and availability of cues. Since the SR HRA-B1 HFEs are from the internal events PRA, the specific procedure guidance and task analysis are contained in the hfe_cr.xls and hfe_cp.xls Excel spreadsheets from the internal events ANO-1 PRA. The HR SRs from Section 2 of the ASME/ANS PRA Standard for the added HFEs have not been performed.	The HRA events have been removed from the "Altered Events" table, except for actions that are set to TRUE in a fire scenario where fire prohibits the operator action. The revised FPRA HRA analysis (PRA-A1-05-008 - ANO-1 Fire PRA Human Failure Events) provides detailed HRA evaluations for most of the fire-specific operator actions. Two of the events are left at screening values (QHFGSGDEPRES = 0.1 and RHF1ESASRG = 1.0). All events used in the FPRA have been developed and documented using the same methods used for the internal events HEPs.	The F&O does not reflect the current model of record. The information below depicts the current modeling and use of a new methodology. The ANO-1 HRA methodology was revised consistent with the approach used to address ANO-2 NFPA 805 LAR RAls. The revised methodology uses the NUREG-1921 methodology with detailed Human Error Probabilities (HEPs) developed for each Human Failure Event (HFE) credited in the FPRA.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		HRA-B3 (continued)	The added HFEs need to be "processed" using the same methods that were employed to develop the HFE for the internals event PRA.		A focused scope peer review was performed in June 2014 evaluating the revised HRA process.	
N/A	Resolved	HRA-C1	<p>Fire PRA Peer Review Finding HRA-C1-01: A review of the fire HFE Evaluation and the recovery rules revealed a discrepancy for one HFE. As shown in the ANO-1-Fire HFE Evaluation spreadsheet, the HFE EHF1DGCRKP had a value of 4.98E-03 in the internal events PRA for ANO-1 with a new value of 2.99E-02 calculated based on the fire conditions. The calculation was reviewed and found to match the fire HEP process. However, when reviewing the recovery rule file, Alrul4p00.txt, the replacement event for EHF1DGCRKP, Z1EHFDGCRK, was found to have the original value 4.98E-03. It was determined that the error was a result of an error when copying the values from one file to another. One error was found in a small sampling so the extent of condition may be larger so may impact the results.</p> <p>Correct the value for Z1EHFDGCRK in Alrul4p00.txt and then review the other "single replacement" values against the new values in the ANO-1-Fire HFE Evaluation spreadsheet.</p>	The value Z1EHFDGCRK was corrected in the FPRA rule recovery file (Alrul4p00_FIRE.txt). The recovery rule file was thoroughly reviewed during the HEP document update to ensure the correct values are applied to the events during recovery.	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	HRA-D2	<p>Fire PRA Peer Review Finding HRA-D2-01: Multiple recovery actions were inserted into the model via the AlteredEvents file. Screening values were used for all of these events so none of them accounted for relevant fire-related effects, including any effects that may preclude a recovery action or alter the manner in which it is accomplished. The values used may be conservative or non-conservative so it is not possible to fully assess the true impact of these recovery actions.</p> <p>ANO-1 plans to determine which of these recovery actions need to be retained after the NFPA 805 Change Evaluation. Once the HFEs to be retained are identified, they need to be fully defined and quantified in accordance with the process used for all the other HFEs.</p>	<p>The HRA events have been removed from the "Altered Events" table, except for actions that are set to TRUE in a fire scenario where fire prohibits the operator action. The revised FPRA HRA analysis (PRA-A1- 05-008 – ANO-1 Fire PRA Human Failure Events) provides sufficient detail to meet all HRA-D2 supporting requirements. These changes include new combinations of operator actions for dependency. All events used in the FPRA have been developed and documented using the same methods used for the internal events HEPs.</p>	See new disposition response for HRA-B3-01	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
N/A	Resolved	HRA-E1	<p>Fire PRA Peer Review Finding HRA-E1-01: In the FRANC Altered events file, there are a number of basic events with replacement values of 0.1. These replacement values represent screening values for Operator Recovery Actions to recover the faulted basis event. This is the only place these "operator actions" show up, they are not proceduralized at this time and they are not documented or evaluated beyond the screening evaluation. At this point in time, these new actions are not proceduralized and are considered to be recovery actions. These actions still need to be evaluated for significance. Entergy has indicated that once these actions have been evaluated, the important ones will be incorporated into procedures and quantified in accordance with their standard process.</p>	<p>The HRA events have been removed from the "Altered Events" table, except for actions that are set to TRUE in a fire scenario where fire prohibits the operator action. As discusses in the response to HRA-B3-01 and HRA-D2-01, these actions are evaluated in detail using the same HRA methodology used in internal events model.</p>	See new disposition response for HRA-B3-01	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		HRA-E1 (continued)	The other actions will be removed. Inclusion of undocumented, unevaluated operator actions in the models can impact the results. Before the fire PRA can be used for applications beyond NFPA 805, the "implied human actions" need to be documented and incorporated in operating procedures. Each such action needs to be clearly identified. Any actions that are not proceduralized need to be removed from the model.			
N/A	Resolved	HRA-E1	Fire PRA Peer Review Finding HRA-E1-02: Report 0247060006.03-U1, Rev. 0 documents the HRA for the ANO-1 Fire PRA. Section 3 documents the assumptions used in the Fire PRA HRA. This section contains a total of 2 assumptions. However, a text search of the report on "assume" found five additional assumptions buried in the text. Another text search on "could" and a text search on "may" yielded another three instances of what appeared to be assumptions. This is considered to be a good indication that not all assumptions have been documented. While capturing all assumptions into a common location may not have a significant impact on the base model, there is a concern for future applications. One step in performing a risk-informed application is to review the assumptions to determine if any of them could impact the application and, if so, what would need to be done to compensate for the assumption if it is nonconservative with respect to the application.	Section 5 of the ANO-1 Fire PRA Human Failure Events report (PRA-A1-05-008) has been expanded and now includes all relevant HRA modeling assumptions. In addition to the general assumptions included in Section 5 of calculation PRA-A1-05-008, each of the detailed post-fire HRA events has assumptions included in the associated evaluation.	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		HRA-E1 (continued)	Review Report 0247060006.03-U1, Rev. 0 to identify additional assumptions and capture them in Section 3. The definition of "assumption" used for the search should be relatively broad so as to capture as many potential assumptions as possible. It is easier to disposition a trivial assumption than it is to address a significant assumption that was not identified as such.			
N/A	Resolved	FQ-A4	<p>Fire PRA Peer Review Finding FQ-A4-01: The uncertainty interval on CDF results was not estimated as required by QU-E3 (LE-F3). An uncertainty analysis has been performed to identify and qualify specific areas of uncertainty. This meets the internal event requirement for QU-E1, QU-E2, and QU-E4.</p> <p>Determine an uncertainty interval based upon the model uncertainties identified in QU-E1 and E2. Provide basis for any non-applicability of any of the requirements under these sections in Part 2.</p>	Calculation PRA-A1-05-006 – ANO-1 Fire PRA Uncertainty/Sensitivity Analysis provides a Monte Carlo evaluation of uncertainty for both the FPRA Core Damage Frequency and the Large Early Release Frequency. The listed uncertainty analysis satisfies the listed Standard requirements (QU-E1, QU-E2, and QU-E3).	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
N/A	Resolved	FQ-A4	<p>Fire PRA Peer Review Finding FQ-A4-02: A spot check of scenario ignition frequencies documented in the FRANC model revealed several errors in the calculations. Based on the number of errors found and the lack of documentation for scenario frequency calculations, this indicates a potentially systemic problem with the scenario ignition frequencies. Review and recalculate scenario ignition frequencies.</p>	The scenario ignition frequencies are calculated in Attachment D of the Fire Scenarios report (PRA-A1-05-004). The calculated scenario ignition frequencies have been verified to be consistent with the calculations performed for the zone frequencies in the Fire Probabilistic Risk Assessment Plant Partitioning and Fire Ignition Frequency Development report (CALC-08-E-0016-01).	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FQ-A4	<p>Fire PRA Peer Review Finding FQ-A4-03: Section 7.4 of Appendix D (MSO Expert Panel Review and Disposition of Open Items) in Report 0247060006.02-U1 describes changes made to the PRA model to account for the potential of an MSO causing a spurious spray event. A review of the model shows that the AND gate FIRE027 that represents this scenario includes an event (RHF1RCPSXP) that is set to 0.0. This will prevent the MSO scenario from being quantified. The model does not accurately quantify an MSO scenario due to a modeling error.</p> <p>Correct the model and review for other potential similar errors.</p>	<p>The modeling error has been corrected. The new event (RHF1RCPSXP) has a default value of 1.0. It is only changed if changed in the altered events table when it is relevant to a specific case. With the default value of 1.0, it does not disrupt quantification of the other logic in the AND gate for other cases.</p>	Original disposition remains valid.	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>
N/A	Resolved	FQ-D1	<p>Fire PRA Peer Review Finding FQ-D1-01: The Fire PRA LERF model is based upon the internal events LERF model. The LERF model uses the Fire PRA plant response model. The frequency for fire-induced LERF is quantified. However, F&Os for element "LE" and other elements were identified in the ANO-1 RG 1.200 peer review of the internal events model, performed in August 2009. These F&Os have not been addressed and limit the LERF modeling capability of the Fire PRA model. Also, comprehensive screening of Interfacing Systems LOCA (ISLOCA) and other significant containment isolation paths for fire scenarios has not been performed. The frequency of different containment failure modes leading to large early release is needed for fire-induced LERF.</p>	<p>F&Os relating to ANO-1 ISLOCA treatment have all been resolved. A revision to the ISLOCA fault tree was performed following the peer review. This revision included an update to the internal events model (and subsequently the FPRA model).</p> <p>The evaluation of containment isolation paths (potential LERF contribution via breaches in containment) are documented in Appendix G of the Component and Cable Selection Report (PRA-A1-05-003).</p> <p>See Table 2 for ISLOCA/LERF findings additional details.</p>	Original disposition remains valid.	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		FQ-D1 (continued)	Resolve internal events model F&Os for element "LE." Comprehensive screening of Interfacing Systems LOCA (ISLOCA) and other potential significant containment failure paths is also needed.			
N/A	Resolved	FQ-E1	<p>Fire PRA Peer Review Finding FQ-E1-01: The ANO-1 Fire PRA results are very conservative for CDF and LERF. There are several important scenarios that are driving the results that have not had detailed modeling performed to reduce the conservatisms. These conservative results may mask other important contributors to the fire risk. This SR requires that significant contributors be identified in accordance with HLR-QU-D. HLR-QU-D6 requires that significant contributors be identified and HLR-QU-D7 requires review of important components and basic events to determine that they make logical sense. This is not possible with overly conservative scenario models. Update model to remove conservatisms.</p>	<p>Since the ANO-1 FPRA Peer Review, ANO has made several refinements within the reviewed methodology to remove some conservatism and reduce overall CDF and LERF. These refinements include:</p> <ul style="list-style-type: none"> • developing more detailed fire scenarios • refining the components failed within the scenario • refining the fire HRA events and JHEPs. <p>Detailed fire modeling has not been applied to ANO fire scenarios based on the limited benefit in CDF and LERF reduction given conservative input parameters.</p>	See new response to FSS-C1-01.	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FQ-E1	<p>Fire PRA Peer Review Finding FQ-E1-02: The ANO-1 Fire PRA Summary Report (ERIN Report 0247060006.06-U1, Rev. 1, 9/30/09) Appendices A and B provide the quantification results for CDF and LERF. Appendix C presents "INSIGHTS / RECOMMENDATIONS (DOMINANT RISK CONTRIBUTORS)." The significant contributors to LERF were not well identified and results were not clearly traced to inputs and assumptions in the Fire PRA. Therefore the SR is not met. The fire-induced LERF quantification results are not reviewed sufficiently to identify significant contributors to LERF. Presentation of dominant LERF risk contributors in Appendix C should be expanded to fully discuss all dominant contributors and their basis in the inputs and assumptions made in the Fire PRA.</p>	<p>The ANO-1 Fire PRA Summary Report (PRA-A1-05-005) has been updated to include additional result details. Attachments D, E, F, G, & H have been added to provide additional details on results.</p> <ul style="list-style-type: none"> Appendix D – Uncertainty and Sensitivity Matrix Appendix E – Cutsets Comprising the Top 90% of CDF Appendix F – CDF Importances Report Appendix G – Cutsets Comprising the Top 90% of LERF Appendix H – LERF Importances Report <p>Section 4.0 of the Summary Report has been updated to explain results, insights, and dominant risk contributors.</p>	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FQ-F1	<p>Fire PRA Peer Review Finding FQ-F1-01: There is no discussion of quantification process limitations as required in QU-F5, or significance definitions (basic event, cutsets, and accident sequences) as required by QU-F6. A discussion of the quantification process limitations is required by QU-F5 by reference through FQ-F1. Also, quantitative definitions for significant basic events, cut sets, and accident sequences are required by QU-F6.</p> <p>Provide the required discussions and definitions and document.</p>	<p>Section 4.2 of the ANO-1 Fire PRA Summary Report (PRA-A1-05-005) discusses the limitations of the PRA software used (CAFTA, FORTE, FRANC). Quantitative results and insights for risk significant sequences are also provided in the scenario report. Significance is defined in Section 4.1 of the report as all scenarios included in 90% of the total Fire CDF/LERF. Tables in the report list risk significant scenarios (4-1 for CDF and 4-2 for LERF), cutsets (Appendix E for CDF and Appendix G for LERF), and basic event importance measures (Appendix F for CDF and Appendix H for LERF).</p>	<p>The original disposition remains valid with the exception that quantitative results and insights for risk significant sequences are not provided in the scenario report.</p>	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>
N/A	Resolved	UNC-A1	<p>Fire PRA Peer Review Finding UNC-A1-01: The uncertainty interval on CDF results was not estimated as required by QU-E3 (LE-F3). An uncertainty analysis has been performed to identify and qualify specific areas of uncertainty. This meets the internal event requirement for HLR-QU-E1, E2, and E4. The uncertainty interval on CDF results was not estimated as required by QU-E3 (LE-F3).</p> <p>Determine an uncertainty interval based upon the model uncertainties identified in QU-E1 and E2. Provide basis for any non-applicability of any of the requirements under these sections in Part 2.</p>	<p>Uncertainty intervals (ones that meet the criteria of QU-E1, E2, and E3) have been developed for both CDF and LERF and are documented in PRA-A1-05-006 – ANO-1 Fire PRA Uncertainty/Sensitivity Analysis.</p> <p>The results of the uncertainty evaluation are also presented in Appendix D of the Summary Report (PRA-A1-05-005).</p>	<p>Original disposition remains valid.</p>	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FSS-G3	<p>Fire PRA Peer Review Finding FSS-G3-01: There is no documented basis for the use of screening criteria value of 5E-7/yr. In addition, given that the threshold value is 1E-7, it is not documented in the report if the screening process has considered cumulative risk.</p> <p>Provide a justification for the screening value of 5E-7. The report should discuss how the screening process deals with cumulative risk.</p>	The updated Multi-Compartment and Hot Gas Layer analysis no longer uses screening criteria for HGL (ZOI) and MCA. This is documented in PRA-A1-05-009 -ANO-1 Multi-Compartment/ Hot Gas Layer Analysis. All zones now include a HGL and multiple MCA scenarios.	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
N/A	Resolved	FSS-G3	<p>Fire PRA Peer Review Finding FSS-G3-04: The assumption of only two cable trays as intervening combustibles may not be conservative or realistic (i.e. reflect as built conditions) for all the PAU's. Given that the multi compartment analysis relies heavily on screening due to hot gas layer conditions in the PAU, accurate intervening combustible input parameters to the fire modeling analysis is necessary.</p> <p>Conduct walkdowns for identifying the correct package of intervening combustibles to use as input to the fire modeling analysis. Walkdowns should be documented.</p>	Walk-downs to identify scenarios with greater than 2 trays were performed and the results have been incorporated into the FPRA (PRA-A1-05-009).	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FSS-G3	<p>Fire PRA Peer Review Finding FSS-G3-05: The use of the manual suppression constant for electrical fires appears to be used for the calculation of all the manual suppression failure probabilities. The manual suppression constant should be applied depending on the specific ignition source/fire that characterizes the fire scenario. In addition, it is not clear in the calculation which time input is used for determining the manual suppression probability values (i.e. at what time hot gas layer is reached?). Based on the response to a question submitted during the peer review process, there are some curves (e.g. the high energy arcing fault) that also could apply to specific PAU's that were not considered.</p> <p>Ensure that the suppression curve selected is bounding and document a technical justification for the selection to be used in the screening process.</p>	The updated ANO-1 Multi-Compartment Analysis (PRA-A1-05-009) contains a discussion of how manual suppression probabilities are determined. This discussion contains the technical details (including justification). The .012 non-suppression curve, when applied, is applied in a bounding condition.	<p>The original disposition to this F&O is not valid for the current modeling. The following response better depicts the current Fire PRA model.</p> <p>The manual non-suppression probability (NSPms) is calculated using a convolution process which applies the above non-suppression curve to each bin of the applicable NUREG/CR-6850, Appendix E heat release rates (HRR), with the time to hot gas layer varying based on the heat release rate associated with each bin. The mean suppression rate from Table 14-1 of NUREG/CR-6850 Supplement 1 for the appropriate fire type is used as opposed to the .0102 bounding value discussed in the original disposition to this F&O.</p>	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FSS-G3	<p>Fire PRA Peer Review Finding FSS-G3-06: There are numerous statements in Table 3-1 under "final disposition" that lack technical justification. Examples include (this F/O is not limited to these examples only):</p> <ul style="list-style-type: none"> "The 0.001 applied is a very conservative factor applied to zones over 353 cu ft. A more appropriate NSPms for this scenario is 1E-04 which results in screening the scenario for MCA impacts" – Question: Where does 0.001 come from? Why it is considered conservative? Where does 1E-4 come from? Why is a factor from FAQ 044 for main feed water pumps (MFWPs) to oil tank rooms used? Are there any other ignition sources other than the pumps in these areas? Given the large volume of the turbine building, no hot gas layer would be able to form which would preclude the MCA impacts – Question: Is this true for MFWP and large turbine generator (TG) fires? Can we preclude HGL scenarios for these ignition sources? 	Table 3-1 (the table in question) has been eliminated. It was removed via the change in screening approach (to address F&O FSS-G3-01). All zones now have an HGL scenario and multiple MCA scenarios and the associated quantification is incorporated into the FPRA model.	Original disposition remains valid.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
		FSS-G3 (continued)	<ul style="list-style-type: none"> The only adjacent zone connected through a door is 2200-MM which will not be impacted due its large volume. Other adjacent zones can use the next worse barrier failure probability for dampers, 0.0027, which lowers the Pmca to 9.94E-08 – Comment: The resulting screening value is barely below the screening criteria. If we are selecting probabilities from a table without considering what is in the boundary in terms of seals and dampers, proper justification for the probabilities are needed. Crediting a 0.02 NSP for emergency diesel generator (EDG) oil fires from Appendix E screens the scenario without crediting manual suppression – Question: Where is this 0.02 coming from? Where is the reference for the fixed system credited? There are tray combustibles within the zone; however they are located 10 ft or more above the floor elevation and would not be impacted by a fire in this zone. Screen the scenario for MCA impacts – Question: Why does the statement "would not be impacted by a fire in this zone" apply to these specifically? <p>Clarify these statements in a way that simplify reviews and future updates of this calculation. Some of these will require clear technical justifications.</p>			

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FSS-G4	<p>Fire PRA Peer Review Finding FSS-G4-01: It is not documented in the multi compartment report how the requirements of this SR were addressed. For example, the standard requires to "CONFIRM that the allowed credit is consistent with the fire-resistance rating as demonstrated by conformance to applicable test standards". There is no evidence that this confirmation has been conducted. Without a systematic process for identifying barrier types between physical analysis units, ANO-1 will need to ensure that addressing this SR will account for all the different barrier types (walls, barriers, spatial separations, doors, etc.). See F/O PP-B3-01.</p> <p>A possible resolution is to list the types of barriers between adjacent PAU's for determining which probabilities are applicable and document if the generic values in Table 11-3 of NUREG/CR-6850 are bounding for ANO-1.</p>	<p>Walk-downs of non-NRC or Insurance commitment fire barriers have been performed to document the basis for credit taken for fire zone boundaries. Quantification of MCA probability has conservatively used the door failure probability from NUREG/CR-6850, Table 11-3, as the boundary failure mechanism for all zones without openings to adjacent fire zones. For zones with openings to adjacent zones, the boundary failure probability was set to 1.0 and the volume of the combined zones was used for assessing the time to HGL formation (PRA-A1-05-009).</p>	See response to PP-C3-01.	<p>This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.</p>

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
N/A	Resolved	FSS-G5	<p>Fire PRA Peer Review Finding FSS-G5-01:</p> <p>It is not documented in the multi compartment report how the requirements of this SR were addressed. For example, the standard requires to "QUANTIFY the effectiveness, reliability, and availability of the active fire barrier element". There is no evidence that this confirmation has been conducted. Without a systematic process for identifying barrier types between physical analysis units, ANO-1 will need to ensure that addressing this SR will account for all the different barrier types (walls, barriers, spatial separations, doors, etc.). See F/O PP-B3-01.</p> <p>A possible resolution is to list the types of barriers between adjacent PAU's for determining which probabilities are applicable and document if the generic values in Table 11-3 of NUREG/CR-6850 are bounding for ANO-1.</p>	<p>Walk-downs of non-NRC or Insurance commitment fire barriers have been performed to document the basis for credit taken for fire zone boundaries. Quantification of MCA probability has conservatively used the door failure probability from NUREG/CR-6850, Table 11-3, as the boundary failure mechanism for all zones without openings to adjacent fire zones. For zones with openings to adjacent zones, the boundary failure probability was set to 1.0 and the volume of the combined zones was used for assessing the time to HGL formation (PRA-A1-05-009).</p>	See response to PP-C3-01.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A1-5136	Resolved	HR-G3 (Finding not submitted in original Att. V of the ANO-1 NFPA 805 LAR)	<p>Fire PRA Peer Review Finding FS HR-G3-01:</p> <p>Observation: Detailed analysis of HPI-HFC-FO-CRSPR-EF and HPI-HFC-FO-CRSPR-IF are two examples where the time available is less than the time required.</p> <p>Basis for Significance: The HCR/ORE quantification for these HFEs are non-conservative and the HEP should be 1.0.</p> <p>Possible Resolution: Set HEPs for HPI-HFC-FO-CRSPR-EF and HPI-HFC-FO-CRSPR-IF to 1.0 or review and update timing information such that time available is greater than time required.</p>	<p>HFEs HPI-HFC-FO-CRSPR-EF/IF were re-examined consistent with the detailed quantification method discussed in Section 7 of PRA-A1-05-015 R0 and timing information was updated such that the time available is greater than the time required for the operator action.</p>	N/A	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
A1-5137	Resolved	HR-G3 (Finding not submitted in original Att. V of the ANO-1 NFPA 805 LAR)	<p>Fire PRA Peer Review Finding FS HR-G3-02: Observation: All assumptions that impact feasibility of operator actions need to be validated before the HEPs can be applied, specifically:</p> <p>Assumptions regarding availability of instrumentation credited for diagnosis should be verified available in the fire scenarios where the HFE is credited. This includes instrumentation credited as diverse to instrumentation that is rendered unavailable due to fire. Note that these assumptions may be implicit in analyses where cues and indications are stated without any statements regarding availability in the fire scenarios where the HFE is credited.</p> <p>Related to the above are assumptions made with regards to explicit modeling of instrumentation. Assumptions regarding procedure changes need to be validated.</p> <p>Basis for Significance: The reliability of operator actions can only be assessed for feasible operator actions. If an operator action is not feasible, it cannot be reliable and the HEP should be 1.0 (refer to NUREG/CR-1921 Sections 3.5 and 4.3 for additional discussion). Risk metrics can be underestimated if feasibility is assumed where operator actions cannot be shown to be feasible.</p> <p>Possible Resolution: Ensure that all assumptions regarding feasibility (whether implicit or explicit) are systematically identified and addressed. It will be necessary to demonstrate this to support the intended NFPA-805 license change request.</p>	<p>The new fire specific HFEs developed in PRA-A1-05-015 R0 were reviewed to identify assumptions related to feasibility that would need to be addressed for NFPA 805; none were identified. [NOTE: feasibility analysis was performed for each HFE that had been identified as risk-significant on the basis of the Fussell-Vesely importance measure. The fire PRA does not take credit for non-risk significant HFEs; these full power internal events (FPIE) HEPs were set to 1.0.]</p> <p>PRA-A1-05-003 R2, Appendix F (Pages F-34 through F-51) show screen shots of the HRA cue instrumentation added. The model change events log (Appendix F) provides the listing of the changes to the fault tree for the FIRE_HRA_1 through FIRE_HRA_18.</p> <p>The original HRA was contained in PRA-A1-05-007 R0 and PRA-A1-05-008 R1 and combined into a single HRA report PRA-A1-05-015 R1 after the Peer review and NFPA-805 RAI responses.</p>	N/A	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements	Importance to Application
A1-5139	Resolved	HR-G7 (Finding not submitted in original Att. V of the ANO-1 NFPA 805 LAR)	<p>Fire PRA Peer Review Finding FS HR-G7-01:</p> <p>Based on a review of Figure 5 in PRA-A2-01-003S03, the dependency approach does not consider availability of resources, which can be important for fire PRA.</p> <p>Basis for Significance:</p> <p>The number of operators required to perform actions implied by a combination of HFEs in a cutset may exceed the number of operators available to perform these actions. As such some actions will not be able to be performed and the HEPs should be 1.0. For standard internal events PRAs, this is not typically an issue, but for fire and other spatial events, the need for more operator actions may produce cutsets with HFEs that are inappropriately credited.</p> <p>Possible Resolution: Review the combinations and determine resources required versus resources available. Justify credit given for HFEs where resources required < resources available e.g. for long enough time windows, additional manpower will be on site to assist.</p>	The dependency analysis performed for the current analysis (PRA-A1-05-015 R0) utilizes the EPRI HRA Calculator, rather than the HRA Toolbox, and therefore addresses the peer review concerns regarding the assessment of dependency factors such as availability of resources.	N/A	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

Table 5

List of SRs Assessed as CC-I in the ANO-1 Fire PRA Model

SR	Topic	Status	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
FSS-C1	Use of Multi-Point Heat Release Rate Treatment	<p>Capability Category 1 is acceptable for the application. While the results are conservative, they are not significantly more so than the also conservative results of more detailed fire modeling.</p> <p>Additionally, some multi-point heat release rate analysis is applied. Section 7.1 of the Fire Scenarios Report outlines the use of a multi-point treatment for vented panels based on a split fraction developed from the EPRI Fire Events Database. This split fraction specifies the fraction of fires impacting only the ignition source panel versus those fires which impact targets within the zone of influence of the panels.</p> <p>Section 16 of the Fire Scenarios Report discusses the use of generic fire modeling versus detailed fire modeling and justifies the ANO approach for the FPRA application.</p>	<p>The original disposition for this F&O does not depict the modeling used in the current MOR.</p> <p>The following information was obtained from the ANO-1 SE (ML16223A481). In PRA RAI 01.c (Reference 21), the NRC staff requested that the licensee explain its approach and whether the split fraction referred to in the resolution to FSS-C1-01 implied that the "Panel Factors method" (Reference 105), not accepted by NRC was used. In its response to PRA RAI 01.c (Reference 10), the licensee clarified that that the "Panel Factors" method was not used but rather generic fire modeling treatments (GFMTs) were used to define a three-point fire model to meet SR FSS-C1.). The licensee explained that the first fire in the three-point fire treatment is a non-severe fire in which the source panel and the cables terminating at the source panel are damaged but the nearest target is not damaged. The second fire in the three-point fire treatment is a severe fire in which all targets within the 98th percentile zone of influence (ZOI) are impacted.</p>	<p>This issue has no impact in STI change evaluations performed in accordance with the SFCP.</p>

SR	Topic	Status	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
FSS-C1 (continued)			The third fire of the three-point fire treatment is a fire that results in a hot gas layer (HGL) exceeding a 80 degree Centigrade (°C) criterion in which all targets in the fire zone are conservatively assumed to be damaged. These three fire models are discussed in the licensee's response to FM RAI 01.f discussed in SE Section 3.4.2.3. The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that it used a multiple-point fire intensity and duration model encompassing low likelihood but risk-contributing fire events consistent with the requirements in the PRA standard.	
PP-B2 (Finding PP-B2-01)	Plant Partitioning – Use of Non-Rated Fire Barriers	<p>In limited PAUs, ANO credits non-rated barriers (spatial separation) and does not full meet Capability Category II. The impacts of these barriers are evaluated (no credit for non-rated barrier) in the scenario development and MCA.</p> <p>SR PP-B2 references NUREG/CR-6850, Chapter 1, for the acceptable criteria for justifying non-rated fire barriers. NUREG/CR-6850 discusses the use of fire compartments as “a well-defined enclosed room, not necessarily with fire barriers.” ANO references the FHA as a starting point for plant partitioning and all barriers (both rated and non-rated are defined in the FHA). The Plant Partitioning Task (CALC-08-E-0016-01, R0) assumes that fire protection features will be effective at containing a fire under most conditions. Fire protection features include fire-rated barriers, non-fire-rated barriers, active features such as water curtains, and in some cases spatial separation. The potential failure of a credited partitioning feature is addressed in the MCA.</p>	Original disposition remains valid.	This issue has no impact in STI change evaluations performed in accordance with the SFCP.

SR	Topic	Status	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
PP-B2 (Finding PP-B2-01) (continued)		The ANO FHA does not include any partitioning features, such as partial height walls, that are discussed in NUREG/CR-6850 as barriers that should not be credited. The adequacy of the fire barriers is explicitly reviewed as part of the Multi-Compartment and Hot Gas Layer Analysis calculation (PRA-A1-05-009).		
PP-B3-01	Plant Partitioning – Use of Spatial Separation	Spatial separation is used in two areas at the site (four total PAUs): the fuel handling area and the turbine deck. Both are very large areas and the development of a hot gas layer is not credited. The spatial separation distance is also sufficiently large such that no fixed or transient sources are capable of impacting the area beyond the separation. While this only meets Capability Category I for partitioning, the follow up tasks (fire scenarios and MCA) show it has no impact on FPRA results or conclusions.	The original F&O disposition is not applicable. The following response better depicts the current Fire PRA model. The current analysis considers spatial separation in a very few areas. In each of these areas, the MCA/HGL analysis assumes no barrier failure probability. Therefore, the significance of the spatial separation is assessed in the analysis results.	This issue has no impact in STI change evaluations performed in accordance with the SFCP.

SR	Topic	Status	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
FSS-B2	Potential fire scenarios leading to the MCR abandonment. (Issue not submitted in original Att. V of the ANO-1 NFPA 805 LAR)	<p>SR was given a CC-I due to the assumptions associated with the MCR abandonment Scenario.</p> <p>The Main Control Room Abandonment analysis uses a bounding approach to characterize fire risk contribution. Section 3.2 of the Fire Scenarios Report (0247-06-0006.05-U1) details MCR abandonment treatment. The scenario is 129-F, Scenario A. A CCDP of 0.1 is used based on an adequate evaluation of Appendix R, III.L requirements. The evaluation is based on adequate alternate shutdown procedures, validation of timing and manual action feasibility. Calculation ANO1-FP-09- 00011, Rev. 2, evaluates the abandonment times for the Unit 1 MCR. These arguments bound the fire risk contribution for MCR abandonment.</p> <p>Additionally, an incorrect calculation reference (Reference 3 in Section 8) of the Fire Scenarios Report (0247-06-0006.05-U1, Rev. 0). The calculation reference is ANO2-FP-09-00011. The correct reference should be ANO1-FP-09-00011.</p>	<p>The determination of MCR abandonment scenario was updated to incorporate the following elements into the analysis; 1) determination of the frequency of abandonment using the multiple ignition sources within the control room, 2) fire protection features, room ventilation and room geometry are considered. 3) NSP values were selected from Calc ANO1-FP-09-00011 assuming multi-cable bundle fire spreads, 4) CCDP was calculated based upon the failures modeled to occur from a fire inside the control room, 5) credit for mitigating operator actions outside the control room. The consideration of these elements in determining the CCDP of the MCR abandonment scenario addressed the bounding assumptions identified during the peer review and provides a realistic assessment of the MCR abandonment scenario.</p> <p>Additionally, the incorrect reference was corrected in subsequent revisions to the Scenario documentation.</p>	This issue has no impact in STI change evaluations performed in accordance with the SFCP.

SR	Topic	Status	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
CF-A2	<p>Characterize the uncertainty associated with the applied conditional failure probability assigned for fire-induced circuit failures</p> <p>(Issue not submitted in original Att. V of the ANO-1 NFPA 805 LAR)</p>	<p>The summary report, Appendix D for task 10 characterizes the uncertainty associated with method employed in determining failure probabilities. The conclusion of this report is that the "application of circuit failure probabilities is considered to have minimal impact on the results." Though it may be that a detailed analysis technique was followed for dominant scenarios, these failures are still in the dominant sequences. The accuracy and uncertainty associated with these values would have a significant impact on the results.</p> <p>Characterize the uncertainty with respect to how the method employed could introduce uncertainty into the final results.</p>	<p>The Uncertainty analysis for the ANO-1 Fire PRA model includes propagation of uncertainty for circuit failure model probabilities. The Hot Short Probabilities (HSP's) were incorporated using a similar approach to the fire ignition frequencies. A type code was developed for each HSP utilized in the ANO-1 model. The mean and its associated variance were assigned based on the information in NUREG/CR-7150. This was done to correlate uncertainty in the HSP's associated with each fire scenario. This approach avoided the inappropriate assumption that the scenario frequencies are independent of each other. Latin Hypercube sampling was performed to propagate parametric uncertainties through the ANO-1 FPRA model to generate probability distributions for ANO-1 fire CDF and LERF.</p>	<p>This issue has no impact in STI change evaluations performed in accordance with the SFCP.</p>

SR	Topic	Status	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
HRA-A4	<p>Talk through (i.e., review in detail) with plant operations and training personnel the procedures and sequence events to confirm that interpretation of the procedures relevant to actions identified in SRs HRA-A1, HRA-A2, and HRA-A3 is consistent with plant operational and training practices.</p> <p>(Issue not submitted in original Att. V of the ANO-1 NFPA 805 LAR)</p>	<p>ANO-1 reviewed the required cues for each identified action against the Appendix R protected instrumentation to confirm that adequate cues would be available. The results of these reviews are documented in Attachment A of the report. Attachment E contains the results of the simulator review study to confirm the availability of the Appendix R cues and the likely actions. These discussions were at a very high/general level and did not address specific procedures.</p> <p>ANO should strengthen the discussions, especially as they pertain to the conclusion that the operators will not take action based on a single indication.</p> <p>ANO should also explicitly identify which specific Appendix R instruments are to be credited for each identified human action.</p>	<p>The original internal events HEPs that were carried over to the FPRA were evaluated in detail through operator interviews and simulator exercise experience as documented in Appendix B of the HRA analysis. Fire specific information used to update these HFEs for the Fire PRA was verified with PRA staff with operations and training experience.</p> <p>As part of detailed analysis of specific events for fire, a feasibility assessment consistent with the guidance in NUREG-1921 was performed. FPIE HFEs were identified as risk-significant on the basis of the F-V importance measure. These risk-significant FPIE HFEs formed the basis for the development of new fire-specific HFEs, which were evaluated for feasibility.</p> <p>The following areas were evaluated to ensure all actions were feasible, and if any action was determined to be non-feasible, a value of 1.0 was assigned to the operator action. The following parameters were evaluated for feasibility: procedures and training, available indications and cues, equipment functionality and accessibility, adequate time available to perform the action, environmental factors, communications, and staffing.</p>	<p>This issue has no impact in STI change evaluations performed in accordance with the SFCP.</p>

SR	Topic	Status	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
HRA-A4 (continued)			Cues and indications are necessary because all required operator actions are preceded by them. Without cues or indications, the operators have no prompts that an action is required and, hence, no operator action can be credited. The analysis must evaluate whether or not the instruments and indications needed for diagnosis are affected by the fire. The current HRA analysis provides information that indicates the Operator Cues and Components Credited with Providing Operator Cues for Post Fire Safe Shutdown for each HFE. Additionally, a table with cues associated with ANO-1 Fire HFEs is provided and documented in the ANO-1 Fire HRA Calculator file.	
HRA-D1	Include operator recovery actions that can restore the functions, systems, or components on an as-needed basis provide a more realistic evaluation.	One recovery action is incorporated into the PRA model for multiple loss of DC breaker for 4160 bus events, which have an accident sequence associated with them. The event found indicates that recovery actions were incorporated for significant sequences rather than universally. The identification of all recovery actions used in the model is documented in Attachment D of ANO-1 Fire HRA Notebook (Report 0247060006.03-U1) Most fire-specific recoveries used screening values so this was set as CC-1.	The ANO-1 HRA methodology was revised consistent with the approach used to address ANO-2 NFPA 805 LAR RAIs. The revised methodology uses the NUREG-1921 methodology with detailed Human Error Probabilities (HEPs) developed for each Human Failure Event (HFE) credited in the FPRA. All events used in the FPRA have been developed and documented using the same methods used for the internal events HEPs.	This issue has no impact in STI change evaluations performed in accordance with the SFCP.

3.4 Identification of Key Assumptions

The Initiative 5b is a risk-informed process that uses PRA model results to support a proposed STI change. The IDP uses the PRA results as an input to decide whether an STI change is warranted. The methodology recognizes that a key area of uncertainty for this application is the standby failure rate utilized in the determination of the STI extension impact. Therefore, the methodology requires the performance of selected sensitivity studies on the standby failure rate of the component(s) of interest for the STI assessment.

Any additional sensitivity studies identified for specific STI changes are also required per NEI 04-10, Revision 1. Therefore, results of the standby failure rate sensitivity study plus the results of any additional sensitivity studies identified during the performance of the reviews of gaps and open items as summarized in Sections 3.2 and 3.3 herein, will be documented and included in the results of the risk analysis submitted to the IDP.

3.5 External Events and Shutdown Considerations

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

External hazards were evaluated in the ANO-1 Individual Plant Examination of External Events (IPEEE) submittal in response to the NRC IPEEE Program (Reference 15). The IPEEE Program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of associated severe accident risks. ANO-1 does not have a PRA model or applications associated with external hazards such as seismic, high wind, or external flooding, and quantitative results cannot be provided to support this STI effort. Therefore, a qualitative or bounding approach will be used to assess external event hazard risk at ANO-1 for STI changes.

Because ANO-1 does not have external hazards or shutdown PRA models, external hazards and shutdown screening evaluations are expected to be performed for STI changes in accordance with the guidance of NEI 04-10, Revision 1.

The ANO-1 shutdown safety program developed to support implementation of NUMARC 91-06 (Reference 16) is used for the shutdown risk evaluation, or an application-specific shutdown analysis may be performed for STI changes in accordance with the guidance of NEI 04-10, Revision 1. The ANO-1 shutdown safety program includes input from a Defense-in-Depth shutdown EOOS PRA model.

4. **CONCLUSIONS**

The information presented herein demonstrate that the ANO-1 PRA technical adequacy and capability evaluations, as well as the maintenance and update processes conform to the ASME/ANS PRA Standard, which satisfies the guidance of RG 1.200, Revision 2.

5. REFERENCES

1. TSTF-425, "Technical Specification Task Force – Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b," Revision 3, March 2009.
2. NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," Revision 1, April 2007.
3. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.
4. ASME RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009.
5. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011.
6. 38-1290272-00, "Arkansas Nuclear One - 1 Probabilistic Risk Assessment Peer Review Report", October 2002.
7. "ANO-1 RG 1.200 PRA Peer Review Report Using ASME PRA Standard Requirements," August 2009.
8. ENTGANO150-REPT-001, "Arkansas Nuclear One Unit 1 Internal Flooding Probabilistic Risk Assessment Peer Review," Revision 0, April 2017.
9. NUREG/CR-6850 – EPRI-1011089, "Fire PRA Methodology for Nuclear Power Facilities," August 2005.
10. PRA-A1-05-005, "ANO-1 Fire PRA Summary Report," Revision 1, 2016.
11. LTR-RAM-II-10-003, "Fire PRA Peer Review Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications for the Arkansas Nuclear One Unit 1 Fire Probabilistic Risk Assessment," January 2010.
12. 0021-0022-005, "Focused Peer Review of the Arkansas Nuclear One Unit 1 Fire Probabilistic Risk Assessment," Kleinsorg Group, May 2012.
13. 5384.R01.121129, "Focused Scope Peer Review ANO-1 Fire PRA FSS-A, C, D, E and H," Kazarians & Associates, Inc., November 2012.
14. "Arkansas Nuclear One Unit 1 Fire HRA Peer Review Report," Curtiss-Wright, June 2014.
15. Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f), Supplement 4," June 1991.
16. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.
17. 1CAN011401, "License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition) Arkansas Nuclear One – Unit 1," January 29, 2014.

ATTACHMENT 3

1CAN031801

PROPOSED TECHNICAL SPECIFICATION CHANGES

1.1 Definition (continued)

SHUTDOWN MARGIN (SDM)	<p>SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:</p> <ol style="list-style-type: none">All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM;In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; andThere is no change in APSR position.
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	<p>THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 The SDM shall be within the limit specified in the COLR.

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM greater than or equal to the limit specified in the COLR.	24 hours In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<div>SR 3.1.2.1</div> <div>-----NOTES-----</div> <div><div>1. The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading.</div><div>2. This Surveillance is not required to be performed prior to entry into MODE 2.</div></div> <div>-----</div> <div>Verify measured core reactivity balance is within $\pm 1\% \Delta k/k$ of predicted values.</div>	<div>Once prior to entering MODE 1 after each fuel loading</div> <div>AND</div> <div>-----NOTE----- Only required after 60 EFPD -----</div> <div>In accordance with the Surveillance Frequency Control Program31 EFPD thereafter</div>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.2.2.3 -----NOTE----- Only required when THERMAL POWER is > 20% RTP. -----</p> <p>Perform SR 3.2.5.1.</p>	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3.	6 hours
C. More than one CONTROL ROD inoperable, or not aligned within 6.5% of its group average height, or both.	<p>C.1.1 Verify SDM to be within the limit provided in the COLR.</p> <p><u>OR</u></p> <p>C.1.2 Initiate boration to restore SDM to within limit.</p> <p><u>AND</u></p> <p>C.2 Be in MODE 3.</p>	<p>1 hour</p> <p>1 hour</p> <p>6 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 Verify individual CONTROL ROD positions are within 6.5% of their group average height.	In accordance with the Surveillance Frequency Control Program12 hours
SR 3.1.4.2 Verify CONTROL ROD freedom of movement for each individual CONTROL ROD that is not fully inserted.	In accordance with the Surveillance Frequency Control Program92 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each safety rod is fully withdrawn.	In accordance with the Surveillance Frequency Control Program12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

LCO 3.1.6 Each APSR shall be OPERABLE and aligned to within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One APSR inoperable, or not aligned to within 6.5% of its group average height, or both.	A.1 Perform SR 3.2.5.1.	2 hours <u>AND</u> 2 hours after each APSR movement
B. Require Action and associated Completion Time not met.	B.1 Be in MODE 3	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.6.1 Verify position of each APSR is within 6.5% of the group average height.	In accordance with the Surveillance Frequency Control Program 12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Position Indicator Channels

LCO 3.1.7 One position indicator channel for each CONTROL ROD and APSR shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

NOTES

Separate Condition entry is allowed for each CONTROL ROD and APSR.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The required position indicator channel inoperable for one or more rods.	A.1 Declare the rod(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 Perform CHANNEL CHECK of required position indicator channel.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.1.7.2 Perform CHANNEL CALIBRATION of required position indicator channel.	In accordance with the Surveillance Frequency Control Program 18 months

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. THERMAL POWER > 85% RTP.</p> <p><u>OR</u></p> <p>Nuclear overpower trip setpoint > 10% higher than PHYSICS TESTS power level.</p> <p><u>OR</u></p> <p>Nuclear overpower trip setpoint > 90% RTP.</p> <p><u>OR</u></p> <p>-----NOTE----- Only required when THERMAL POWER is > 20% RTP. -----</p> <p>LHR not within limits.</p>	B.1 Suspend PHYSICS TESTS exceptions.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify THERMAL POWER is \leq 85% RTP.	In accordance with the Surveillance Frequency Control Program 1 hour
SR 3.1.8.2	<p>-----NOTE----- Only required when THERMAL POWER is > 20% RTP. -----</p> <p>Perform SR 3.2.5.1.</p>	In accordance with the Surveillance Frequency Control Program 2 hours
SR 3.1.8.3	Verify nuclear overpower trip setpoint \leq 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP.	Within 8 hours prior to performance of PHYSICS TESTS at each test plateau

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Moved from Page 3.1.8-2

SURVEILLANCE		FREQUENCY
SR 3.1.8.2	<p>-----NOTE----- Only required when THERMAL POWER is > 20% RTP. -----</p> <p>Perform SR 3.2.5.1.</p>	In accordance with the Surveillance Frequency Control Program 2 hours
SR 3.1.8.3	Verify nuclear overpower trip setpoint \leq 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP.	Within 8 hours prior to performance of PHYSICS TESTS at each test plateau
SR 3.1.8.4	Verify SDM to be within the limits provided in the COLR.	In accordance with the Surveillance Frequency Control Program 24 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Nuclear overpower trip setpoint is not within limit.</p> <p><u>OR</u></p> <p>Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit inoperable.</p>	C.1 Suspend PHYSICS TESTS exceptions.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.9.1 Verify THERMAL POWER is $\leq 5\%$ RTP.	In accordance with the Surveillance Frequency Control Program 1 hour
SR 3.1.9.2 Verify nuclear overpower trip setpoint is $\leq 5\%$ RTP.	Within 8 hours prior to performance of PHYSICS TESTS
SR 3.1.9.3 Verify SDM to be within the limit provided in the COLR.	In accordance with the Surveillance Frequency Control Program 24 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Regulating rod groups inserted in unacceptable operation region.	D.1 Initiate boration to restore SDM to within the limit provided in the COLR.	15 minutes
	<u>AND</u>	
	D.2.1 Restore regulating rod groups to within restricted operation region.	2 hours
	<u>OR</u>	
	D.2.2 Reduce THERMAL POWER to less than or equal to the THERMAL POWER allowed by the regulating rod group insertion limits.	2 hours
E. Required Actions and associated Completion Times of Conditions C or D not met.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	In accordance with the Surveillance Frequency Control Program12 hours
SR 3.2.1.2	Verify regulating rod groups meet the insertion limits as specified in the COLR.	In accordance with the Surveillance Frequency Control Program12 hours
SR 3.2.1.3	Verify $SDM \geq 1\% \Delta k/k$.	Within 4 hours prior to achieving criticality

3.2 POWER DISTRIBUTION LIMITS

3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

LCO 3.2.2 APSRs shall be positioned within the limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. APSRs not within limits.	A.1 -----NOTE----- Only required when THERMAL POWER is > 20% RTP. ----- Perform SR 3.2.5.1. <u>AND</u>	Once per 2 hours
	A.2 Restore APSRs to within limits.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify APSRs are within acceptable limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program 12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL POWER IMBALANCE Operating Limits

LCO 3.2.3 AXIAL POWER IMBALANCE shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 40% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AXIAL POWER IMBALANCE not within limits.	A.1 Perform SR 3.2.5.1.	Once per 2 hours
	<u>AND</u> A.2 Reduce AXIAL POWER IMBALANCE to within limits.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to \leq 40% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify AXIAL POWER IMBALANCE is within limits as specified in the COLR.	In accordance with the Surveillance Frequency Control Program 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.4.1 Verify QPT is within limits as specified in the COLR.	<p>In accordance with the Surveillance Frequency Control Program 7 days</p> <p><u>AND</u></p> <p>When QPT has been restored to less than or equal to the steady state limit, 1 hour for 12 consecutive hours, or until verified acceptable at $\geq 95\%$ RTP</p>

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action C.1 and referenced in Table 3.3.1-1.	D.1 Be in MODE 3.	6 hours
	<u>AND</u> D.2 Open all control rod drive (CRD) trip breakers.	6 hours
E. As required by Required Action C.1 and referenced in Table 3.3.1-1.	E.1 Open all CRD trip breakers.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.1-1.	F.1 Reduce THERMAL POWER < 45% RTP.	6 hours
G. As required by Required Action C.1 and referenced in Table 3.3.1-1.	G.1 Reduce THERMAL POWER < 10% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply to each RPS Function.

SURVEILLANCE	FREQUENCY
SR 3.3.1.1 Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program12 hours

SURVEILLANCE	FREQUENCY
<p>SR 3.3.1.2</p> <p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Adjust power range channel output if the absolute difference is $> 2\%$ RTP. 2. Not required to be performed until 24 hours after THERMAL POWER is $\geq 20\%$ RTP. <p>-----</p> <p>Compare results of calorimetric heat balance calculation to power range channel output.</p>	<p>In accordance with the Surveillance Frequency Control Program 96 hours</p> <p><u>AND</u></p> <p>Once within 24 hours after a THERMAL POWER change of $\geq 10\%$ RTP</p>
<p>SR 3.3.1.3</p> <p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Adjust the power range channel imbalance output if the absolute value of the imbalance error is $\geq 2\%$ RTP. 2. Not required to be performed until 24 hours after THERMAL POWER is $\geq 20\%$ RTP. <p>-----</p> <p>Compare results of out of core measured AXIAL POWER IMBALANCE to incore measured AXIAL POWER IMBALANCE.</p>	<p>In accordance with the Surveillance Frequency Control Program 31 days</p>
<p>SR 3.3.1.4</p> <p>Perform CHANNEL FUNCTIONAL TEST.</p>	<p>In accordance with the Surveillance Frequency Control Program 31 days</p>

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SURVEILLANCE		FREQUENCY
SR 3.3.1.5	<div>-----NOTE-----</div> <div>Neutron detectors are excluded from CHANNEL CALIBRATION.</div> <div>-----</div> <div>Perform CHANNEL CALIBRATION.</div>	<div>In accordance with the Surveillance Frequency Control Program</div> <div>18 months</div>

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SURVEILLANCE		FREQUENCY
SR 3.3.1.5	<div>-----NOTE-----</div> <div>Neutron detectors are excluded from CHANNEL CALIBRATION.</div> <div>-----</div> <div>Perform CHANNEL CALIBRATION.</div>	<div>In accordance with the Surveillance Frequency Control Program</div> <div>18 months</div>

Table 3.3.1-1 Moved to Next (new) Page

Table 3.3.1-1
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower – a. High Setpoint	1,2 ^(a) ,3 ^(d)	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.4 SR 3.3.1.5	≤ 104.9% RTP
b. Low Setpoint	2 ^(b) ,3 ^(b) 4 ^(b) ,5 ^(b)	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 618 °F
3. RCS High Pressure	1,2 ^(a) ,3 ^(d)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 2355 psig
4. RCS Low Pressure	1,2 ^(a)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 1800 psig
5. RCS Variable Low Pressure	1,2 ^(a)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
6. Reactor Building High Pressure	1,2,3 ^(c)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 18.7 psia
7. Reactor Coolant Pump to Power	1,2 ^(a)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 55% RTP with one pump operating in each loop.
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE	1,2 ^(a)	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
9. Main Turbine Trip (Oil Pressure)	≥ 45% RTP	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 40.5 psig
10. Loss of Main Feedwater Pumps (Control Oil Pressure)	≥ 10% RTP	G	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 55.5 psig
11. Shutdown Bypass RCS High Pressure	2 ^(b) ,3 ^(b) 4 ^(b) ,5 ^(b)	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 1720 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) With any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

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CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two or more RTMs inoperable in MODE 4 or 5. <u>OR</u> Required Action and associated Completion Time not met in MODE 4 or 5.	C.1 Open all CRD trip breakers. <u>OR</u> C.2 Remove power from all CRD trip breakers.	6 hours 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.3.1 Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program 92 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program92 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.3.5.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.3.5.3	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program18 months

3.3 INSTRUMENTATION

3.3.7 Engineered Safeguards Actuation System (ESAS) Actuation Logic

LCO 3.3.7 The ESAS digital actuation logic channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguards equipment is
required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each digital actuation logic channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more digital actuation logic channels inoperable.	A.1 Place associated component(s) in engineered safeguards configuration.	1 hour
	<u>OR</u> A.2 Declare the associated component(s) inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform digital actuation logic CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program 31 days

3.3 INSTRUMENTATION

3.3.8 Diesel Generator (DG) Loss of Power Start (LOPS)

LCO 3.3.8 Two loss of voltage Function relays and two degraded voltage Function relays DG LOPS instrumentation per DG shall be OPERABLE.

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

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more relays for one or more DGs inoperable.	A.1 Restore relay(s) to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Declare affected DG(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.8.1 Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program 7 days

SURVEILLANCE	FREQUENCY
<p>SR 3.3.8.2</p> <p>-----NOTE-----</p> <p>When DG LOPS instrumentation is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed up to 4 hours for the loss of voltage Function, provided the one remaining relay monitoring the Function for the bus is OPERABLE.</p> <p>-----</p> <p>Perform CHANNEL CALIBRATION with setpoint Allowable Value as follows:</p> <ul style="list-style-type: none"> a. Degraded voltage ≥ 423.2 V and ≤ 436.0 V with a time delay of 8 seconds ± 1 second; and b. Loss of voltage ≥ 1600 V and ≤ 3000 V with a time delay of ≥ 0.30 seconds and ≤ 0.98 seconds. 	<p>In accordance with the Surveillance Frequency Control Program 18 months</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.3.9.2	<p>-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. -----</p> <p>Perform CHANNEL CALIBRATION.</p>	In accordance with the Surveillance Frequency Control Program 18 months

3.3 INSTRUMENTATION

3.3.10 Intermediate Range Neutron Flux

LCO 3.3.10 One intermediate range neutron flux channel shall be OPERABLE.

APPLICABILITY: MODE 2
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required channel inoperable.	-----NOTE----- Plant temperature changes are allowed provided the temperature change is accounted for in the SDM calculations. -----	Immediately
	A.1 Suspend operations involving positive reactivity changes.	1 hour
	<u>AND</u> A.2 Open CRD trip breakers.	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.10.1 Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.3.10.2 Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program 31 days

SURVEILLANCE		FREQUENCY
SR 3.3.10.3	<div>-----NOTE-----</div> <div>Neutron detectors are excluded from CHANNEL CALIBRATION.</div> <div>-----</div> <div>Perform CHANNEL CALIBRATION.</div>	<div>In accordance with the Surveillance Frequency Control Program</div> <div>18 months</div>

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met for Function 1.a or 1.d.	E.1 Reduce THERMAL POWER to $\leq 10\%$ RTP.	6 hours
F. Required Action and associated Completion Time not met for Functions 1.c, 2, or 3.	F.1 Be in MODE 3. <u>AND</u> F.2 Reduce steam generator pressure to < 750 psig.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.11-1 to determine which SRs shall be performed for each EFIC Function.

SURVEILLANCE	FREQUENCY
SR 3.3.11.1 Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.3.11.2 Perform CHANNEL FUNCTIONAL TEST. (Notes 1 & 2)	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.3.11.3 Perform CHANNEL CALIBRATION. (Notes 1 & 2)	In accordance with the Surveillance Frequency Control Program 18 months

Moved to Page 3.3.11-3

The following notes apply only to the SG Level – Low function:

- Note 1: If the as-found channel setpoints are conservative with respect to the Allowable Value but outside their predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoints are not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
- Note 2: The instrument channel setpoint(s) shall be reset to a value that is equal to or more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint and the predefined as-found acceptance criteria band are specified in the Bases.

SURVEILLANCE REQUIREMENTS (continued)

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The following notes apply only to the SG Level – Low function:

Note 1: If the as-found channel setpoints are conservative with respect to the Allowable Value but outside their predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoints are not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Note 2: The instrument channel setpoint(s) shall be reset to a value that is equal to or more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint and the predefined as-found acceptance criteria band are specified in the Bases.

Table 3.3.11-1
Emergency Feedwater Initiation and Control System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUES
1. EFW Initiation				
a. Loss of MFW Pumps (Control Oil Pressure)	≥ 10% RTP	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 55.5 psig
b. SG Level - Low	1,2,3	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 9.34 inches ^(c,d)
c. SG Pressure - Low	1,2,3 ^(a)	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig
d. RCP Status	≥ 10% RTP	4	SR 3.3.11.1 SR 3.3.11.2	NA
2. EFW Vector Valve Control				
a. SG Pressure – Low	1,2,3 ^(a)	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig
b. SG Differential Pressure – High	1,2,3 ^(a)	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≤ 150 psid
3. Main Steam Line Isolation				
a. SG Pressure – Low	1,2,3 ^{(a)(b)}	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig

(a) When SG pressure ≥ 750 psig.

(b) Except when all associated valves are closed and deactivated.

(c) The SG Level – Low “Limiting Trip Setpoint” in accordance with NRC letter dated September 7, 2005, *Technical Specification For Addressing Issues Related To Setpoint Allowable Values*, is ≥ 10.42 inches.

(d) Includes an actuation time delay of ≤ 10.4 seconds.

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CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met for EFW Initiation Function.	D.1 Be in MODE 3. <u>AND</u>	6 hours
	D.2 Be in MODE 4.	12 hours
E. Required Action and associated Completion Time not met for Main Steam Line Isolation Function.	E.1 Be in MODE 3. <u>AND</u>	6 hours
	E.2.1 Reduce steam generator pressure to < 750 psig. <u>OR</u>	12 hours
	E.2.2 Close and deactivate all associated valves.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.12.1 Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program 31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.13.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program31 days

3.3 INSTRUMENTATION

3.3.14 Emergency Feedwater Initiation and Control (EFIC) Vector Logic.

LCO 3.3.14 Four channels of the EFIC vector logic shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 when steam generator pressure is ≥ 750 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One vector logic channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 Reduce steam generator pressure to < 750 psig.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.14.1 Perform a CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program 31 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action D.1 and referenced in Table 3.3.15-1.	E.1 Be in MODE 3.	6 hours
	<u>AND</u> E.2 Be in MODE 4.	12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.15-1.	F.1 Initiate action to prepare and submit a Special Report.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
These SRs apply to each PAM instrumentation Function in Table 3.3.15-1.

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.3.15.2	<p>-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. -----</p> <p>Perform CHANNEL CALIBRATION.</p>	In accordance with the Surveillance Frequency Control Program 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.16.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.3.16.2	<p>-----NOTE-----</p> <p>When the Control Room Isolation – High Radiation instrumentation is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 3 hours.</p> <p>-----</p> <p>Perform CHANNEL FUNCTIONAL TEST.</p>	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.3.16.3	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program 18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters (loop pressure, hot leg temperature, and RCS total flow rate) shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

-----NOTE-----
RCS loop pressure limit does not apply during pressure transients due to a THERMAL POWER change > 5% RTP per minute.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.1.1 -----NOTE----- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. ----- Verify RCS loop pressure is within the limit specified in the COLR.</p>	<p>In accordance with the Surveillance Frequency Control Program 12 hours</p>

SURVEILLANCE		FREQUENCY
SR 3.4.1.2	<p>-----NOTE----- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. -----</p> <p>Verify RCS hot leg temperature is within the limit specified in the COLR.</p>	<p>In accordance with the Surveillance Frequency Control Program 12 hours</p>
SR 3.4.1.3	<p>Verify RCS total flow is within the limit specified in the COLR.</p>	<p>In accordance with the Surveillance Frequency Control Program 12 hours</p>
SR 3.4.1.4	<p>-----NOTE----- Only required to be performed when stable thermal conditions are established at $\geq 90\%$ RTP. -----</p> <p>Verify RCS total flow rate is within the limit specified in the COLR by measurement.</p>	<p>In accordance with the Surveillance Frequency Control Program 18 months</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 The RCS average temperature (T_{avg}) shall be $\geq 525\text{ }^{\circ}\text{F}$.

APPLICABILITY: MODE 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} not within limit.	A.1 Be in MODE 3.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS $T_{avg} \geq 525\text{ }^{\circ}\text{F}$.	In accordance with the Surveillance Frequency Control Program12 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. -----NOTE----- Required Action D.2 shall be completed whenever this Condition is entered. ----- Requirements of LCO not met in other than MODE 1, 2, 3, or 4.	D.1 Initiate action to restore parameter(s) to within limit. <u>AND</u> D.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	<p>-----NOTE-----</p> <p>Only required to be performed during RCS heatup operations with fuel in the reactor vessel.</p> <p>-----</p> <p>Verify RCS pressure, RCS temperature, and RCS heatup rates are within the limits specified in Figure 3.4.3-1.</p>	<p>In accordance with the Surveillance Frequency Control Program</p> <p>30 minutes</p>
SR 3.4.3.2	<p>-----NOTE-----</p> <p>Only required to be performed during RCS cooldown operations with fuel in the reactor vessel.</p> <p>-----</p> <p>Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-2.</p>	<p>In accordance with the Surveillance Frequency Control Program</p> <p>30 minutes</p>
SR 3.4.3.3	<p>-----NOTE-----</p> <p>Only required to be performed during RCS heatup and cooldown operations with no fuel in the reactor vessel.</p> <p>-----</p> <p>Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-3.</p>	<p>In accordance with the Surveillance Frequency Control Program</p> <p>30 minutes</p>

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SURVEILLANCE		FREQUENCY
SR 3.4.3.3	-----NOTE----- Only required to be performed during RCS heatup and cooldown operations with no fuel in the reactor vessel. -----	In accordance with the Surveillance Frequency Control Program 30 minutes
	Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-3.	
SR 3.4.3.4	-----NOTE----- Only required to be performed during PHYSICS TESTS with RCS temperature ≤ 525 °F. -----	In accordance with the Surveillance Frequency Control Program 30 minutes
	Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.3-1.	

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops – MODES 1 and 2

- LCO 3.4.4 Two RCS Loops shall be in operation, with:
- Four reactor coolant pumps (RCPs) operating; or
 - Three RCPs operating and THERMAL POWER restricted as specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCP not in operation in each loop.	A.1 Restore one non-operating RCP to operation.	18 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> LCO not met for reasons other than Condition A.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify required RCS loops are in operation.	In accordance with the Surveillance Frequency Control Program 12 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two RCS loops inoperable. <u>OR</u> Required RCS loop not in operation.	C.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u> C.2 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loop is in operation.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.4.5.2	<p>-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. -----</p> <p>Verify correct breaker alignment and indicated power available to each required pump.</p>	In accordance with the Surveillance Frequency Control Program 7 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two required loops inoperable. <u>OR</u> Required loop not in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u> B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify required DHR or RCS loop is in operation.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.4.6.2	-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. ----- Verify correct breaker alignment and indicated power available to each required pump.	In accordance with the Surveillance Frequency Control Program 7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify required DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.4.7.2	Verify required SG secondary side water levels are ≥ 20 inches.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.4.7.3	<p>-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. -----</p> <p>Verify correct breaker alignment and indicated power available to each required DHR pump.</p>	In accordance with the Surveillance Frequency Control Program 7 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No required DHR loop OPERABLE. <u>OR</u> Required DHR loop not in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u> B.2 Suspend all operations involving reduction in RCS water volume.	Immediately
	<u>AND</u> B.3 Initiate action to restore one DHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify required DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.4.8.2	<p>-----NOTE-----</p> <p>Not required to be performed until 24 hours after a required pump is not in operation.</p> <p>-----</p> <p>Verify correct breaker alignment and indicated power available to each required DHR pump.</p>	In accordance with the Surveillance Frequency Control Program 7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level \leq 320 inches.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.4.9.2	Verify capacity of ES bus powered pressurizer heaters \geq 126 kW.	In accordance with the Surveillance Frequency Control Program 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Verify pressurizer level does not represent a water solid condition.	30 minutes during RCS heatup and cooldown <u>AND</u> In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.4.11.2	Verify HPI is deactivated.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.4.11.3	Verify each pressurized CFT is isolated.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.4.11.4	<p>-----NOTE-----</p> <p>Verification of locked, sealed, or otherwise secured open vent path(s) only required to be performed every 31 days.</p> <p>-----</p> <p>Verify OPERABLE pressure relief capability.</p>	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.4.11.5	Perform CHANNEL CALIBRATION of ERV opening circuitry.	In accordance with the Surveillance Frequency Control Program 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	<p>-----NOTE-----</p> <p>Only required to be performed in MODE 1 and 2, MODE 3 with RCS average temperature ≥ 500 °F.</p> <p>-----</p> <p>Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 2200 $\mu\text{Ci/gm}$.</p>	In accordance with the Surveillance Frequency Control Program 7 days
SR 3.4.12.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 $\mu\text{Ci/gm}$.	In accordance with the Surveillance Frequency Control Program 14 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.13.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Not required to be performed until 12 hours after establishment of steady state operation at or near operating pressure. 2. Not applicable to primary to secondary LEAKAGE. <p>-----</p> <p>Verify RCS operational LEAKAGE is within limits by performance of an RCS water inventory balance.</p>	<p>In accordance with the Surveillance Frequency Control Program 72 hours</p>
<p>SR 3.4.13.2 -----NOTE-----</p> <p>Not required to be performed until 12 hours after establishment of steady state operation.</p> <p>-----</p> <p>Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.</p>	<p>In accordance with the Surveillance Frequency Control Program 72 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY										
SR 3.4.14.1	<p>-----NOTE----- Not required to be performed in MODES 3 and 4. -----</p> <p>Verify leakage from each RCS pressure isolation check valve, or pair of check valves, as applicable, is less than or equal to an equivalent of the Allowable Leakage Limit identified below at a differential test pressure ≥ 150 psid.</p> <table><thead><tr><th><u>Pressure Isolation Check Valve(s)</u></th><th><u>Allowable Leakage Limit</u></th></tr></thead><tbody><tr><td>DH-14A</td><td>≤ 5 gpm</td></tr><tr><td>DH-13A and DH-17</td><td>≤ 5 gpm total</td></tr><tr><td>DH-14B</td><td>≤ 5 gpm</td></tr><tr><td>DH-13B and DH-18</td><td>≤ 5 gpm total</td></tr></tbody></table>	<u>Pressure Isolation Check Valve(s)</u>	<u>Allowable Leakage Limit</u>	DH-14A	≤ 5 gpm	DH-13A and DH-17	≤ 5 gpm total	DH-14B	≤ 5 gpm	DH-13B and DH-18	≤ 5 gpm total	<p>In accordance with the INSERVICE TESTING PROGRAM</p> <p><u>AND</u></p> <p>Once prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months</p>
<u>Pressure Isolation Check Valve(s)</u>	<u>Allowable Leakage Limit</u>											
DH-14A	≤ 5 gpm											
DH-13A and DH-17	≤ 5 gpm total											
DH-14B	≤ 5 gpm											
DH-13B and DH-18	≤ 5 gpm total											
SR 3.4.14.2	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual high RCS pressure signal.	In accordance with the Surveillance Frequency Control Program 18 months										
SR 3.4.14.3	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual high RCS pressure signal: a. ≤ 340 psig for one valve; and b. ≤ 400 psig for the other valve.	In accordance with the Surveillance Frequency Control Program 18 months										
SR 3.4.14.4	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual Core Flood Tank isolation valve “not closed” signal.	In accordance with the Surveillance Frequency Control Program 18 months										
SR 3.4.14.5	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual Core Flood Tank isolation valve “not closed” signal.	In accordance with the Surveillance Frequency Control Program 18 months										

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of required reactor building atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of required reactor building atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program 92 days
SR 3.4.15.3	Perform CHANNEL CALIBRATION of required reactor building atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program 18 months
SR 3.4.15.4	Perform CHANNEL CALIBRATION of required reactor building sump monitor.	In accordance with the Surveillance Frequency Control Program 18 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Core Flood Tanks (CFTs)

LCO 3.5.1 Two CFTs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > 800 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CFT inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One CFT inoperable for reasons other than Condition A.	B.1 Restore CFT to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Two CFTs inoperable.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce RCS pressure to ≤ 800 psig.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.1.1 Verify each CFT isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.5.1.2 Verify borated water volume in each CFT is $\geq 970 \text{ ft}^3$ and $\leq 1110 \text{ ft}^3$.	In accordance with the Surveillance Frequency Control Program 12 hours

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Moved to Page 3.5.1-2

SURVEILLANCE		FREQUENCY
SR 3.5.1.3	Verify nitrogen cover pressure in each CFT is ≥ 560 psig and ≤ 640 psig.	In accordance with the Surveillance Frequency Control Program 12 hours

SURVEILLANCE		FREQUENCY	
Moved from Page 3.5.1-1	SR 3.5.1.2	Verify borated water volume in each CFT is $\geq 970 \text{ ft}^3$ and $\leq 1110 \text{ ft}^3$.	In accordance with the Surveillance Frequency Control Program12 hours
	SR 3.5.1.3	Verify nitrogen cover pressure in each CFT is $\geq 560 \text{ psig}$ and $\leq 640 \text{ psig}$.	In accordance with the Surveillance Frequency Control Program12 hours
	SR 3.5.1.4	Verify boron concentration in each CFT is $\geq 2270 \text{ ppm}$.	In accordance with the Surveillance Frequency Control Program31 days
			AND -----NOTE----- Only required to be performed for affected CFT ----- Once within 12 hours after each solution level increase of ≥ 0.2 feet that is not the result of addition from a borated water source of known concentration $\geq 2270 \text{ ppm}$
	SR 3.5.1.5	Verify power is removed from each CFT isolation valve operator.	In accordance with the Surveillance Frequency Control Program31 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) temperature > 350 °F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 Reduce RCS temperature to ≤ 350 °F.	12 hours
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	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.	In accordance with the Surveillance Frequency Control Program 34 days
SR 3.5.2.2	Verify each ECCS 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

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Moved from Page 3.5.1-1

SURVEILLANCE		FREQUENCY
SR 3.5.2.2	Verify each ECCS 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 3.5.2.3	Verify each ECCS 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.	In accordance with the Surveillance Frequency Control Program 18 months
SR 3.5.2.4	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program 18 months
SR 3.5.2.5	Verify, by visual inspection, each ECCS train reactor building sump suction inlet is not restricted by debris and screens show no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	<p>-----NOTE-----</p> <p>Only required to be performed when ambient air temperature is < 40 °F or > 110 °F.</p> <p>-----</p> <p>Verify BWST borated water temperature is ≥ 40 °F and ≤ 110 °F.</p>	<p>In accordance with the Surveillance Frequency Control Program 24 hours</p>
SR 3.5.4.2	<p>Verify BWST borated water level is ≥ 38.4 feet and ≤ 42 feet.</p>	<p>In accordance with the Surveillance Frequency Control Program 7 days</p>
SR 3.5.4.3	<p>Verify BWST boron concentration is ≥ 2270 ppm and ≤ 2670 ppm.</p>	<p>In accordance with the Surveillance Frequency Control Program 7 days</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	<p>-----NOTE-----</p> <ol style="list-style-type: none"> 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. <p>-----</p> <p>Perform required air lock leakage rate testing in accordance with the Reactor Building Leakage Rate Testing Program.</p>	In accordance with the Reactor Building Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	In accordance with the Surveillance Frequency Control Program 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each reactor building purge isolation valve is closed.	In accordance with the Surveillance Frequency Control Program 34 days
SR 3.6.3.2	<p>-----NOTE----- Valves and blind flanges in high radiation areas may be verified by use of administrative means. -----</p> <p>Verify each reactor building isolation manual valve and blind flange that is located outside the reactor building and not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.</p>	In accordance with the Surveillance Frequency Control Program 34 days
SR 3.6.3.3	<p>-----NOTE----- Valves and blind flanges in high radiation areas may be verified by use of administrative means. -----</p> <p>Verify each reactor building isolation manual valve and blind flange that is located inside the reactor building and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.</p>	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 reactor building isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.5	Verify each automatic reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program 18 months

3.6 REACTOR BUILDING SYSTEMS

3.6.4 Reactor Building Pressure

LCO 3.6.4 Reactor building pressure shall be ≥ -1.0 psig and $\leq +3.0$ psig.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building pressure not within limits.	A.1 Restore reactor building pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 -----NOTE----- LCO 3.0.4.a is not applicable when entering Mode 4. ----- Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify reactor building pressure is ≥ -1.0 psig and $\leq +3.0$ psig.	In accordance with the Surveillance Frequency Control Program 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify each reactor building spray manual, power operated, and automatic valve in each required flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.6.5.2	Operate each required reactor building cooling train fan unit for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.6.5.3	Verify each required reactor building cooling train cooling water flow rate is ≥ 1200 gpm.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.6.5.4	Verify each required reactor building 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 3.6.5.5	Verify each automatic reactor building spray valve in each required flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program 18 months
SR 3.6.5.6	Verify each required reactor building spray pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program 18 months
SR 3.6.5.7	Verify each required reactor building cooling train starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program 18 months
SR 3.6.5.8	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage

3.6 REACTOR BUILDING SYSTEMS

3.6.6 Spray Additive System

LCO 3.6.6 The Spray Additive System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each Spray Additive System 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.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.6.6.2	Verify sodium hydroxide tank solution volume is ≥ 9000 gallons.	In accordance with the Surveillance Frequency Control Program 184 days
SR 3.6.6.3	Verify sodium hydroxide tank solution concentration is > 6.0 wt% and < 8.5 wt.% NaOH.	In accordance with the Surveillance Frequency Control Program 184 days
SR 3.6.6.4	Verify each Spray Additive System automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program 18 months

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SURVEILLANCE		FREQUENCY
SR 3.6.6.3	Verify sodium hydroxide tank solution concentration is > 6.0 wt% and < 8.5 wt.% NaOH.	In accordance with the Surveillance Frequency Control Program 184 days
SR 3.6.6.4	Verify each Spray Additive System automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	<p>-----NOTE-----</p> <p>Only required to be performed in MODES 1 and 2.</p> <p>-----</p> <p>Verify isolation time of each MSIV is within the limits specified in the INSERVICE TESTING PROGRAM.</p>	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.2.2	<p>-----NOTE-----</p> <ol style="list-style-type: none"> Only required to be performed in MODES 1 and 2. Not required to be met when SG pressure is < 750 psig. <p>-----</p> <p>Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.</p>	In accordance with the Surveillance Frequency Control Program 18 months

MFIVs, Main Feedwater Block Valves,
Low Load Feedwater Control Valves and
Startup Feedwater Control Valves
3.7.3

SURVEILLANCE		FREQUENCY
SR 3.7.3.2	-----NOTES-----	
	1. Only required to be performed in MODES 1 and 2.	
	2. Not required to be met when SG pressure is < 750 psig.	
	Verify that each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program 18 months

3.7 PLANT SYSTEMS

3.7.4 Secondary Specific Activity

LCO 3.7.4 The specific activity of the secondary coolant shall be $\leq 0.1 \mu\text{Ci/gm}$ 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.	6 hours
	<u>AND</u> A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.4.1 Verify the specific activity of the secondary coolant is $\leq 0.1 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.	In accordance with the Surveillance Frequency Control Program 31 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Be in MODE 4.	18 hours
D. Two EFW trains inoperable in MODE 1, 2, or 3.	D.1 -----NOTE----- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. ----- Initiate action to restore one EFW train to OPERABLE status.	Immediately
E. Required EFW train inoperable in MODE 4.	E.1 Initiate action to restore EFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program 34 days
SR 3.7.5.2	-----NOTE----- Not required to be performed for the turbine driven EFW pump, until 24 hours after reaching ≥ 750 psig in the steam generators. ----- Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE		FREQUENCY
SR 3.7.5.3	<p>-----NOTE----- Not required to be met in MODE 4 when steam generator is relied upon for heat removal. -----</p> <p>Verify each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.</p>	<p>In accordance with the Surveillance Frequency Control Program 18 months</p>
SR 3.7.5.4	<p>-----NOTE----- Not required to be met in MODE 4 when steam generator is relied upon for heat removal. -----</p> <p>Verify each EFW pump starts automatically on an actual or simulated actuation signal.</p>	<p>In accordance with the Surveillance Frequency Control Program 18 months</p>
SR 3.7.5.5	Verify proper alignment of the required EFW flow paths by verifying manual valve alignment from the "Q" condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever the unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days
SR 3.7.5.6	Verify that feedwater is delivered to each steam generator using the motor-driven EFW pump.	<p>In accordance with the Surveillance Frequency Control Program 18 months</p>

3.7 PLANT SYSTEMS

3.7.6 Q Condensate Storage Tank (QCST)

LCO 3.7.6 The QCST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The QCST inoperable.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours <u>AND</u> Once per 12 hours thereafter
	<u>AND</u> A.2 Restore QCST to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4 without reliance on steam generator for heat removal.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 Verify QCST volume is $\geq 267,000$ gallons when required for both units and $\geq 107,000$ gallons when only required for Unit 1.	In accordance with the Surveillance Frequency Control Program 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	<p>-----NOTE-----</p> <p>Isolation of SWS flow to individual components does not render the SWS inoperable.</p> <p>-----</p> <p>Verify each SWS 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.</p>	<p>In accordance with the Surveillance Frequency Control Program 31 days</p>
SR 3.7.7.2	<p>Verify each SWS 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.</p>	<p>In accordance with the Surveillance Frequency Control Program 18 months</p>
SR 3.7.7.3	<p>Verify each required SWS pump starts automatically on an actual or simulated signal.</p>	<p>In accordance with the Surveillance Frequency Control Program 18 months</p>

3.7 PLANT SYSTEMS

3.7.8 Emergency Cooling Pond (ECP)

LCO 3.7.8 The ECP shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Degradation of the ECP noted pursuant to SR 3.7.8.4 below or by other inspection.	A.1 Determine ECP remains acceptable for continued operation.	7 days
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> LCO not met for reasons other than Condition A.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.8.1 Verify that the indicated water level of the ECP is greater than or equal to that required for an ECP volume of 70 acre-ft.	In accordance with the Surveillance Frequency Control Program24 hours
SR 3.7.8.2 -----NOTE----- Only required to be performed from June 1 through September 30. ----- Verify average water temperature is ≤ 100 °F.	In accordance with the Surveillance Frequency Control Program24 hours

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SURVEILLANCE		FREQUENCY
SR 3.7.8.2	<p>-----NOTE-----</p> <p>Only required to be performed from June 1 through September 30.</p> <p>-----</p> <p>Verify average water temperature is ≤ 100 °F.</p>	In accordance with the Surveillance Frequency Control Program 24 hours
SR 3.7.8.3	<p>Perform soundings of the ECP to verify:</p> <ol style="list-style-type: none"> 1. A contained water volume of ECP ≥ 70 acre-feet, and 2. The minimum indicated water level needed to ensure a volume of 70 acre-feet is maintained. 	In accordance with the Surveillance Frequency Control Program 12 months
SR 3.7.8.4	Perform visual inspection of the ECP to verify conformance with design requirements.	In accordance with the Surveillance Frequency Control Program 12 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate each CREVS train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.7.9.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.9.3	Verify the CREVS automatically isolates the Control Room and switches into a recirculation mode of operation on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program 18 months
SR 3.7.9.4	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 3.7.10.1	Verify each CREACS train starts, operates for at least 1 hour, and maintains control room air temperature ≤ 84 °F D. B.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.7.10.2	Verify system flow rate of 9900 cfm \pm 10%.	In accordance with the Surveillance Frequency Control Program 18 months

3.7 PLANT SYSTEMS

3.7.11 Penetration Room Ventilation System (PRVS)

LCO 3.7.11 Two PRVS trains shall be OPERABLE.

-----NOTE-----
The penetration room negative pressure boundary may be opened intermittently under administrative controls.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PRVS train inoperable.	A.1 Restore PRVS train to OPERABLE status.	7 days
B. Two PRVS trains inoperable due to inoperable penetration room negative pressure boundary.	B.1 Restore penetration room negative pressure boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met. <u>OR</u> Both PRVS trains inoperable for reasons other than Condition B.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.11.1 Operate each PRVS train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program 34 days
SR 3.7.11.2 Perform required PRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

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SURVEILLANCE REQUIREMENTS		
SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Operate each PRVS train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.7.11.2	Perform required PRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SURVEILLANCE		FREQUENCY
SR 3.7.11.3	Verify each PRVS train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program 18 months

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool Water Level.

LCO 3.7.13 The spent fuel pool water level shall be \geq 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool water level not within limit.	A.1 -----NOTE----- LCO 3.0.3 is not applicable. ----- Suspend movement of irradiated fuel assemblies in the spent fuel pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify the spent fuel pool water level is \geq 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	In accordance with the Surveillance Frequency Control Program7 days

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Boron Concentration

LCO 3.7.14 The spent fuel pool boron concentration shall be > 2000 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

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 spent fuel pool.	Immediately
	<u>AND</u> A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the spent fuel pool boron concentration is > 2000 ppm.	In accordance with the Surveillance Frequency Control Program 7 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. <u>AND</u>	6 hours
	F.2 -----NOTE----- LCO 3.0.4.a is not applicable when entering Mode 4. ----- Be in MODE 4.	12 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	In accordance with the Surveillance Frequency Control Program 7 days
SR 3.8.1.2	-----NOTE----- All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. ----- Verify each DG starts from standby conditions and, in ≤ 15 seconds achieves "ready-to-load" conditions.	In accordance with the Surveillance Frequency Control Program 31 days

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and follow, without shutdown, a successful performance of SR 3.8.1.2. <p>-----</p> <p>Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2475 kW and ≤ 2750 kW.</p>	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.8.1.4	Verify each day tank contains ≥ 160 gallons of fuel oil.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.8.1.6	Verify the fuel oil transfer system operates to transfer fuel oil from storage tanks to the day tank.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.8.1.7	<p>-----NOTE-----</p> <p>This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced.</p> <p>-----</p>	

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SURVEILLANCE	FREQUENCY
Verify automatic transfer of AC power sources to the selected offsite circuit and manual transfer to the alternate required offsite circuit.	In accordance with the Surveillance Frequency Control Program18 months

SURVEILLANCE	FREQUENCY
<div data-bbox="87 405 120 737" data-label="Text" style="writing-mode: vertical-rl; transform: rotate(180deg);">Moved from Page 3.8.1-5</div> <div data-bbox="207 367 355 396">SR 3.8.1.7</div> <div data-bbox="440 367 1109 567"> <p>-----NOTE-----</p> <p>This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced.</p> <p>-----</p> <p>Verify automatic transfer of AC power sources to the selected offsite circuit and manual transfer to the alternate required offsite circuit.</p> </div>	<div data-bbox="1149 636 1403 768" data-label="Text"> <p>In accordance with the Surveillance Frequency Control Program 18 months</p> </div>
<div data-bbox="207 840 355 869">SR 3.8.1.8</div> <div data-bbox="440 840 1109 936"> <p>-----NOTE-----</p> <p>All DG starts may be preceded by an engine prelube period.</p> <p>-----</p> <p>Verify on an actual or simulated loss of offsite power signal:</p> </div> <div data-bbox="440 1089 1109 1535"> <ol style="list-style-type: none"> a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: <ol style="list-style-type: none"> 1. achieves "ready-to-load" conditions in ≤ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected shutdown load through automatic load sequencing timers, and 4. supplies connected loads for ≥ 5 minutes. </div>	<div data-bbox="1149 999 1403 1131" data-label="Text"> <p>In accordance with the Surveillance Frequency Control Program 18 months</p> </div>
<div data-bbox="87 1581 120 1938" data-label="Text" style="writing-mode: vertical-rl; transform: rotate(180deg);">Moved to new Page 3.8.1-7</div> <div data-bbox="207 1608 355 1638">SR 3.8.1.9</div> <div data-bbox="440 1608 1109 1705"> <p>-----NOTE-----</p> <p>All DG starts may be preceded by an engine prelube period.</p> <p>-----</p> <p>Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal:</p> </div> <div data-bbox="440 1890 985 1919"> <ol style="list-style-type: none"> a. De-energization of emergency buses; </div>	<div data-bbox="1149 1766 1403 1898" data-label="Text"> <p>In accordance with the Surveillance Frequency Control Program 18 months</p> </div>

Moved to new Page 3.8.1-7	<table><tr><td data-bbox="146 210 1133 642"><ul style="list-style-type: none">b. Load shedding from emergency buses; andc. DG auto-starts from standby condition and:<ul style="list-style-type: none">1. achieves “ready-to-load” conditions in ≤ 15 seconds,2. energizes permanently connected loads,3. energizes auto-connected emergency loads through load sequencing timers, and4. supplies connected loads for ≥ 5 minutes.</td><td data-bbox="1133 210 1430 642"></td></tr></table>	<ul style="list-style-type: none">b. Load shedding from emergency buses; andc. DG auto-starts from standby condition and:<ul style="list-style-type: none">1. achieves “ready-to-load” conditions in ≤ 15 seconds,2. energizes permanently connected loads,3. energizes auto-connected emergency loads through load sequencing timers, and4. supplies connected loads for ≥ 5 minutes.	
<ul style="list-style-type: none">b. Load shedding from emergency buses; andc. DG auto-starts from standby condition and:<ul style="list-style-type: none">1. achieves “ready-to-load” conditions in ≤ 15 seconds,2. energizes permanently connected loads,3. energizes auto-connected emergency loads through load sequencing timers, and4. supplies connected loads for ≥ 5 minutes.			

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SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.9</p> <p>-----NOTE----- All DG starts may be preceded by an engine prelube period. -----</p> <p>Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal:</p> <ol style="list-style-type: none"> De-energization of emergency buses; Load shedding from emergency buses; and DG auto-starts from standby condition and: <ol style="list-style-type: none"> achieves “ready-to-load” conditions in ≤ 15 seconds, energizes permanently connected loads, energizes auto-connected emergency loads through load sequencing timers, and supplies connected loads for ≥ 5 minutes. 	<p>In accordance with the Surveillance Frequency Control Program 18 months</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p data-bbox="211 369 354 394">SR 3.8.2.1</p> <div data-bbox="438 369 1112 558"> <p data-bbox="725 369 831 394">-----NOTES-----</p> <ol data-bbox="440 405 1110 535" style="list-style-type: none"> <li data-bbox="440 405 1110 434">1. SR 3.8.1.3 is not required to be performed. <li data-bbox="440 472 1110 535">2. The 15 second acceptance criteria of SR 3.8.1.2 is not applicable. <p data-bbox="438 606 1112 735">-----</p> <p data-bbox="438 606 1112 735">For AC Sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources – Operating," except SR 3.8.1.4, SR 3.8.1.7, SR 3.8.1.8, and SR 3.8.1.9, are applicable.</p> </div>	<p data-bbox="1153 606 1395 735">In accordance with the Surveillance Frequency Control Program 31 days</p>

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. Required Action and associated Completion Time not met.</p> <p><u>OR</u></p> <p>One or more DGs with diesel fuel oil or required starting air subsystem not within limits for reasons other than Condition A, B, C, or D.</p>	<p>E.1 Declare associated DG inoperable.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains $\geq 20,000$ gallons of fuel.	In accordance with the Surveillance Frequency Control Program 31 days
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
SR 3.8.3.3	Verify each DG required air start receiver pressure is ≥ 175 psig.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.8.3.4	Check for and remove accumulated water from each fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program 31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.4.1 Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	In accordance with the Surveillance Frequency Control Program 7 days
SR 3.8.4.2 Verify each battery charger supplies ≥ 300 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours. <u>OR</u> Verify each battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	In accordance with the Surveillance Frequency Control Program 18 months
SR 3.8.4.3 -----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. ----- 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 or a modified performance discharge test.	In accordance with the Surveillance Frequency Control Program 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	<p>-----NOTE-----</p> <p>Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1.</p> <p>-----</p> <p>Verify each battery float current is ≤ 2 amps.</p>	In accordance with the Surveillance Frequency Control Program 7 days
SR 3.8.6.2	Verify each battery pilot cell float voltage is ≥ 2.07 V.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.8.6.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.8.6.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	In accordance with the Surveillance Frequency Control Program 31 days
SR 3.8.6.5	Verify each battery connected cell float voltage is ≥ 2.07 V.	In accordance with the Surveillance Frequency Control Program 92 days
SR 3.8.6.6	<p>-----NOTE-----</p> <p>This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR.</p> <p>-----</p> <p>Verify battery capacity is $\geq 80\%$ of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.</p>	In accordance with the Surveillance Frequency Control Program 60 months

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Moved to new Page 3.8.6-4	<div></div> <div><u>AND</u> 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating</div>
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SURVEILLANCE	FREQUENCY
<div>SR 3.8.6.6</div> <div>-----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. -----</div> <div>Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.</div>	<div>In accordance with the Surveillance Frequency Control Program60 months</div> <div>AND</div> <div>12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating</div> <div>AND</div> <div>24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating</div>

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 -----NOTE----- LCO 3.0.4.a is not applicable when entering Mode 4. ----- Be in MODE 4.	12 hours
C. Two or more of the four inverters required by LCO 3.8.7.a and LCO 3.8.7.b inoperable.	C.1 Be in MODE 3. <u>AND</u>	12 hours
	C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to associated 120 VAC buses RS1, RS2, RS3, and RS4.	In accordance with the Surveillance Frequency Control Program 7 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.5 Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by AC vital bus inverter(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.8.1 Verify correct inverter voltage and alignments to required 120 VAC vital buses.	In accordance with the Surveillance Frequency Control Program 7 days

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 REQUIREMENTS		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments to required AC, DC, and 120 VAC bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program 7 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.2.4 Initiate actions to restore required AC, DC, and 120 VAC vital bus electrical power distribution subsystems to OPERABLE status.	Immediately
	<u>AND</u>	
	A.2.5 Declare associated required decay heat removal subsystem(s) inoperable.	Immediately
	<u>AND</u>	
	A.2.6 Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by Electrical Power Distribution System.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments to required AC, DC, and 120 VAC vital bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program7 days

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

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

APPLICABILITY: MODE 6.

-----NOTE-----
Only applicable to the refueling canal when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	A.3 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.	In accordance with the Surveillance Frequency Control Program 72 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.9.2.2	<p>-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. -----</p> <p>Perform CHANNEL CALIBRATION.</p>	In accordance with the Surveillance Frequency Control Program 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required reactor building penetration is in the required status.	In accordance with the Surveillance Frequency Control Program 7 days
SR 3.9.3.2	<p>-----NOTE-----</p> <p>Not required to be met for reactor building isolation valves and reactor building purge isolation valves in penetrations closed to comply with LCO c.1.</p> <p>-----</p> <p>Verify each required reactor building isolation valve and each reactor building purge isolation valve actuates to the isolation position.</p>	In accordance with the Surveillance Frequency Control Program 18 months
SR 3.9.3.3	Perform CHANNEL CALIBRATION of reactor building purge exhaust radiation monitor.	In accordance with the Surveillance Frequency Control Program 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program12 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No DHR loop OPERABLE or in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	<u>AND</u>	
	B.2 Initiate action to restore one DHR loop to OPERABLE status and to operation.	Immediately
	<u>AND</u>	
	B.3 Close all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program 12 hours
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to each required DHR pump.	In accordance with the Surveillance Frequency Control Program 7 days

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

LCO 3.9.6 Refueling canal water level shall be maintained \geq 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of irradiated fuel assemblies within the reactor building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within the reactor building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling canal water level is \geq 23 feet above the top of irradiated fuel assemblies seated within the reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program24 hours

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The program shall include the following:

- a. Preventive maintenance and periodic visual inspection requirements; and
- b. Integrated leak test requirements for each system at ~~least once per 18 months~~ a frequency in accordance with the Surveillance Frequency Control Program. The provisions of SR 3.0.2 are applicable.

5.5.3 Iodine Monitoring

This program provides controls that ensure the capability to accurately determine the airborne iodine concentration under accident conditions. The program shall include the following:

- a. Training of personnel;
- b. Procedures for monitoring; and
- c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;
- b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming 10 CFR 20, Appendix B, Table II, Column 2;

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.5 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

- a. The definition of the CRE and the CRE boundary.
- b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.
- c. Requirements for (i) determining the unfiltered air leakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," 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. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency ~~of 18 months on a STAGGERED TEST BASIS~~ in accordance with the Surveillance Frequency Control Program. The results shall be trended and used as part of the ~~18-month assessment of the~~ CRE boundary assessment specified in TS 5.5.5.c.
- e. The quantitative limits on unfiltered air leakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air leakage measured by the testing described in paragraph c. The unfiltered air leakage limit for radiological challenges is the leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air leakage 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 leakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel. Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program inspection frequencies.

5.5.8 ~~DELETED~~ Surveillance Frequency Control Program

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

- a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
- b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
- c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
 1. an API gravity or an absolute specific gravity within limits,
 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
 3. water and sediment within limits;
- b. Within 31 days following addition of new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil;
- c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested ~~every 31 days~~ based on ASTM D-2276, Method A-2 or A-3 [at a Frequency in accordance with the Surveillance Frequency Control Program](#); and
- d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance Frequencies.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
 1. A change in the TS incorporated in the license; or
 2. A change to the updated SAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

Proposed changes that do meet these criteria shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the SAR.

ATTACHMENT 4

1CAN031801

REVISED TECHNICAL SPECIFICATION PAGES

1.1 Definition (continued)

SHUTDOWN MARGIN (SDM)	<p>SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:</p> <ol style="list-style-type: none">All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM;In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; andThere is no change in APSR position.
THERMAL POWER	<p>THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 The SDM shall be within the limit specified in the COLR.

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM greater than or equal to the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<div>SR 3.1.2.1</div> <div>-----NOTES-----</div> <div><div>1. The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading.</div><div>2. This Surveillance is not required to be performed prior to entry into MODE 2.</div></div> <div>-----</div> <div>Verify measured core reactivity balance is within $\pm 1\% \Delta k/k$ of predicted values.</div>	<div>Once prior to entering MODE 1 after each fuel loading</div> <div><u>AND</u></div> <div>-----NOTE-----</div> <div>Only required after 60 EFPD</div> <div>-----</div> <div>In accordance with the Surveillance Frequency Control Program</div>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.2.2.3 -----NOTE----- Only required when THERMAL POWER is > 20% RTP. -----</p> <p>Perform SR 3.2.5.1.</p>	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3.	6 hours
C. More than one CONTROL ROD inoperable, or not aligned within 6.5% of its group average height, or both.	<p>C.1.1 Verify SDM to be within the limit provided in the COLR.</p> <p><u>OR</u></p> <p>C.1.2 Initiate boration to restore SDM to within limit.</p> <p><u>AND</u></p> <p>C.2 Be in MODE 3.</p>	<p>1 hour</p> <p>1 hour</p> <p>6 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 Verify individual CONTROL ROD positions are within 6.5% of their group average height.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.2 Verify CONTROL ROD freedom of movement for each individual CONTROL ROD that is not fully inserted.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each safety rod is fully withdrawn.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

LCO 3.1.6 Each APSR shall be OPERABLE and aligned to within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One APSR inoperable, or not aligned to within 6.5% of its group average height, or both.	A.1 Perform SR 3.2.5.1.	2 hours <u>AND</u> 2 hours after each APSR movement
B. Require Action and associated Completion Time not met.	B.1 Be in MODE 3	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.6.1 Verify position of each APSR is within 6.5% of the group average height.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Position Indicator Channels

LCO 3.1.7 One position indicator channel for each CONTROL ROD and APSR shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

NOTES

Separate Condition entry is allowed for each CONTROL ROD and APSR.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The required position indicator channel inoperable for one or more rods.	A.1 Declare the rod(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Perform CHANNEL CHECK of required position indicator channel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.2	Perform CHANNEL CALIBRATION of required position indicator channel.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. THERMAL POWER > 85% RTP.</p> <p><u>OR</u></p> <p>Nuclear overpower trip setpoint > 10% higher than PHYSICS TESTS power level.</p> <p><u>OR</u></p> <p>Nuclear overpower trip setpoint > 90% RTP.</p> <p><u>OR</u></p> <p>-----NOTE----- Only required when THERMAL POWER is > 20% RTP. -----</p> <p>LHR not within limits.</p>	<p>B.1 Suspend PHYSICS TESTS exceptions.</p>	<p>1 hour</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.8.1 Verify THERMAL POWER is \leq 85% RTP.</p>	<p>In accordance with the Surveillance Frequency Control Program</p>

SURVEILLANCE		FREQUENCY
SR 3.1.8.2	<p>-----NOTE----- Only required when THERMAL POWER is > 20% RTP. -----</p> <p>Perform SR 3.2.5.1.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.1.8.3	Verify nuclear overpower trip setpoint \leq 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP.	Within 8 hours prior to performance of PHYSICS TESTS at each test plateau
SR 3.1.8.4	Verify SDM to be within the limits provided in the COLR.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Nuclear overpower trip setpoint is not within limit.</p> <p><u>OR</u></p> <p>Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit inoperable.</p>	C.1 Suspend PHYSICS TESTS exceptions.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify THERMAL POWER is $\leq 5\%$ RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.1.9.2	Verify nuclear overpower trip setpoint is $\leq 5\%$ RTP.	Within 8 hours prior to performance of PHYSICS TESTS
SR 3.1.9.3	Verify SDM to be within the limit provided in the COLR.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Regulating rod groups inserted in unacceptable operation region.	D.1 Initiate boration to restore SDM to within the limit provided in the COLR.	15 minutes
	<u>AND</u>	
	D.2.1 Restore regulating rod groups to within restricted operation region.	2 hours
	<u>OR</u>	
	D.2.2 Reduce THERMAL POWER to less than or equal to the THERMAL POWER allowed by the regulating rod group insertion limits.	2 hours
E. Required Actions and associated Completion Times of Conditions C or D not met.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.2.1.2	Verify regulating rod groups meet the insertion limits as specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.2.1.3	Verify $SDM \geq 1\% \Delta k/k$.	Within 4 hours prior to achieving criticality

3.2 POWER DISTRIBUTION LIMITS

3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

LCO 3.2.2 APSRs shall be positioned within the limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. APSRs not within limits.	A.1 -----NOTE----- Only required when THERMAL POWER is > 20% RTP. ----- Perform SR 3.2.5.1. <u>AND</u>	Once per 2 hours
	A.2 Restore APSRs to within limits.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify APSRs are within acceptable limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL POWER IMBALANCE Operating Limits

LCO 3.2.3 AXIAL POWER IMBALANCE shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 40% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AXIAL POWER IMBALANCE not within limits.	A.1 Perform SR 3.2.5.1.	Once per 2 hours
	<u>AND</u> A.2 Reduce AXIAL POWER IMBALANCE to within limits.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to \leq 40% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify AXIAL POWER IMBALANCE is within limits as specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.4.1 Verify QPT is within limits as specified in the COLR.</p>	<p>In accordance with the Surveillance Frequency Control Program</p> <p><u>AND</u></p> <p>When QPT has been restored to less than or equal to the steady state limit, 1 hour for 12 consecutive hours, or until verified acceptable at $\geq 95\%$ RTP</p>

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action C.1 and referenced in Table 3.3.1-1.	D.1 Be in MODE 3.	6 hours
	<u>AND</u> D.2 Open all control rod drive (CRD) trip breakers.	6 hours
E. As required by Required Action C.1 and referenced in Table 3.3.1-1.	E.1 Open all CRD trip breakers.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.1-1.	F.1 Reduce THERMAL POWER < 45% RTP.	6 hours
G. As required by Required Action C.1 and referenced in Table 3.3.1-1.	G.1 Reduce THERMAL POWER < 10% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply to each RPS Function.

SURVEILLANCE	FREQUENCY
SR 3.3.1.1 Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE	FREQUENCY
<p>SR 3.3.1.2</p> <p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Adjust power range channel output if the absolute difference is $> 2\%$ RTP. 2. Not required to be performed until 24 hours after THERMAL POWER is $\geq 20\%$ RTP. <p>-----</p> <p>Compare results of calorimetric heat balance calculation to power range channel output.</p>	<p>In accordance with the Surveillance Frequency Control Program</p> <p><u>AND</u></p> <p>Once within 24 hours after a THERMAL POWER change of $\geq 10\%$ RTP</p>
<p>SR 3.3.1.3</p> <p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Adjust the power range channel imbalance output if the absolute value of the imbalance error is $\geq 2\%$ RTP. 2. Not required to be performed until 24 hours after THERMAL POWER is $\geq 20\%$ RTP. <p>-----</p> <p>Compare results of out of core measured AXIAL POWER IMBALANCE to incore measured AXIAL POWER IMBALANCE.</p>	<p>In accordance with the Surveillance Frequency Control Program</p>
<p>SR 3.3.1.4</p> <p>Perform CHANNEL FUNCTIONAL TEST.</p>	<p>In accordance with the Surveillance Frequency Control Program</p>

SURVEILLANCE		FREQUENCY
SR 3.3.1.5	<div>-----NOTE-----</div> <div>Neutron detectors are excluded from CHANNEL CALIBRATION.</div> <div>-----</div> <div>Perform CHANNEL CALIBRATION.</div>	In accordance with the Surveillance Frequency Control Program

Table 3.3.1-1
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower – a. High Setpoint	1,2 ^(a) ,3 ^(d)	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.4 SR 3.3.1.5	≤ 104.9% RTP
b. Low Setpoint	2 ^(b) ,3 ^(b) 4 ^(b) ,5 ^(b)	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 618 °F
3. RCS High Pressure	1,2 ^(a) ,3 ^(d)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 2355 psig
4. RCS Low Pressure	1,2 ^(a)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 1800 psig
5. RCS Variable Low Pressure	1,2 ^(a)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
6. Reactor Building High Pressure	1,2,3 ^(c)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 18.7 psia
7. Reactor Coolant Pump to Power	1,2 ^(a)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 55% RTP with one pump operating in each loop.
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE	1,2 ^(a)	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
9. Main Turbine Trip (Oil Pressure)	≥ 45% RTP	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 40.5 psig
10. Loss of Main Feedwater Pumps (Control Oil Pressure)	≥ 10% RTP	G	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 55.5 psig
11. Shutdown Bypass RCS High Pressure	2 ^(b) ,3 ^(b) 4 ^(b) ,5 ^(b)	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 1720 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) With any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two or more RTMs inoperable in MODE 4 or 5. <u>OR</u> Required Action and associated Completion Time not met in MODE 4 or 5.	C.1 Open all CRD trip breakers. <u>OR</u> C.2 Remove power from all CRD trip breakers.	6 hours 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.3.1 Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.3	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.7 Engineered Safeguards Actuation System (ESAS) Actuation Logic

LCO 3.3.7 The ESAS digital actuation logic channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguards equipment is
required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each digital actuation logic channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more digital actuation logic channels inoperable.	A.1 Place associated component(s) in engineered safeguards configuration.	1 hour
	<u>OR</u> A.2 Declare the associated component(s) inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform digital actuation logic CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.8 Diesel Generator (DG) Loss of Power Start (LOPS)

LCO 3.3.8 Two loss of voltage Function relays and two degraded voltage Function relays DG LOPS instrumentation per DG shall be OPERABLE.

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

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more relays for one or more DGs inoperable.	A.1 Restore relay(s) to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Declare affected DG(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.8.1 Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE	FREQUENCY
<p>SR 3.3.8.2</p> <p>-----NOTE-----</p> <p>When DG LOPS instrumentation is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed up to 4 hours for the loss of voltage Function, provided the one remaining relay monitoring the Function for the bus is OPERABLE.</p> <p>-----</p> <p>Perform CHANNEL CALIBRATION with setpoint Allowable Value as follows:</p> <ul style="list-style-type: none"> a. Degraded voltage ≥ 423.2 V and ≤ 436.0 V with a time delay of 8 seconds ± 1 second; and b. Loss of voltage ≥ 1600 V and ≤ 3000 V with a time delay of ≥ 0.30 seconds and ≤ 0.98 seconds. 	<p>In accordance with the Surveillance Frequency Control Program</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.9.2	<div>-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.</div>	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.10 Intermediate Range Neutron Flux

LCO 3.3.10 One intermediate range neutron flux channel shall be OPERABLE.

APPLICABILITY: MODE 2
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required channel inoperable.	-----NOTE----- Plant temperature changes are allowed provided the temperature change is accounted for in the SDM calculations. -----	
	A.1 Suspend operations involving positive reactivity changes.	Immediately
	<u>AND</u> A.2 Open CRD trip breakers.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.10.1 Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.10.2 Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.3.10.3	<div>-----NOTE-----</div> <div>Neutron detectors are excluded from CHANNEL CALIBRATION.</div> <div>-----</div> <div>Perform CHANNEL CALIBRATION.</div>	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met for Function 1.a or 1.d.	E.1 Reduce THERMAL POWER to $\leq 10\%$ RTP.	6 hours
F. Required Action and associated Completion Time not met for Functions 1.c, 2, or 3.	F.1 Be in MODE 3. <u>AND</u> F.2 Reduce steam generator pressure to < 750 psig.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.11-1 to determine which SRs shall be performed for each EFIC Function.

SURVEILLANCE	FREQUENCY
SR 3.3.11.1 Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.11.2 Perform CHANNEL FUNCTIONAL TEST. ^(Notes 1 & 2)	In accordance with the Surveillance Frequency Control Program
SR 3.3.11.3 Perform CHANNEL CALIBRATION. ^(Notes 1 & 2)	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

The following notes apply only to the SG Level – Low function:

- Note 1: If the as-found channel setpoints are conservative with respect to the Allowable Value but outside their predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoints are not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
- Note 2: The instrument channel setpoint(s) shall be reset to a value that is equal to or more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint and the predefined as-found acceptance criteria band are specified in the Bases.

Table 3.3.11-1
Emergency Feedwater Initiation and Control System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUES
1. EFW Initiation				
a. Loss of MFW Pumps (Control Oil Pressure)	≥ 10% RTP	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 55.5 psig
b. SG Level - Low	1,2,3	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 9.34 inches ^(c,d)
c. SG Pressure - Low	1,2,3 ^(a)	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig
d. RCP Status	≥ 10% RTP	4	SR 3.3.11.1 SR 3.3.11.2	NA
2. EFW Vector Valve Control				
a. SG Pressure – Low	1,2,3 ^(a)	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig
b. SG Differential Pressure – High	1,2,3 ^(a)	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≤ 150 psid
3. Main Steam Line Isolation				
a. SG Pressure – Low	1,2,3 ^{(a)(b)}	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ 584.2 psig

(a) When SG pressure ≥ 750 psig.

(b) Except when all associated valves are closed and deactivated.

(c) The SG Level – Low “Limiting Trip Setpoint” in accordance with NRC letter dated September 7, 2005, *Technical Specification For Addressing Issues Related To Setpoint Allowable Values*, is ≥ 10.42 inches.

(d) Includes an actuation time delay of ≤ 10.4 seconds.

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met for EFW Initiation Function.	D.1 Be in MODE 3. <u>AND</u>	6 hours
	D.2 Be in MODE 4.	12 hours
E. Required Action and associated Completion Time not met for Main Steam Line Isolation Function.	E.1 Be in MODE 3. <u>AND</u>	6 hours
	E.2.1 Reduce steam generator pressure to < 750 psig. <u>OR</u>	12 hours
	E.2.2 Close and deactivate all associated valves.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.12.1 Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.13.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.14 Emergency Feedwater Initiation and Control (EFIC) Vector Logic.

LCO 3.3.14 Four channels of the EFIC vector logic shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 when steam generator pressure is ≥ 750 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One vector logic channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 Reduce steam generator pressure to < 750 psig.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.14.1 Perform a CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action D.1 and referenced in Table 3.3.15-1.	E.1 Be in MODE 3.	6 hours
	<u>AND</u> E.2 Be in MODE 4.	12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.15-1.	F.1 Initiate action to prepare and submit a Special Report.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
These SRs apply to each PAM instrumentation Function in Table 3.3.15-1.

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	In accordance with the Surveillance Frequency Control Program
SR 3.3.15.2	<p>-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. -----</p> <p>Perform CHANNEL CALIBRATION.</p>	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.16.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.16.2	<p>-----NOTE-----</p> <p>When the Control Room Isolation – High Radiation instrumentation is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 3 hours.</p> <p>-----</p> <p>Perform CHANNEL FUNCTIONAL TEST.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.3.16.3	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters (loop pressure, hot leg temperature, and RCS total flow rate) shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

-----NOTE-----
RCS loop pressure limit does not apply during pressure transients due to a THERMAL POWER change > 5% RTP per minute.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.1.1 -----NOTE----- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. ----- Verify RCS loop pressure is within the limit specified in the COLR.</p>	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.4.1.2	<p>-----NOTE----- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. -----</p> <p>Verify RCS hot leg temperature is within the limit specified in the COLR.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.3	Verify RCS total flow is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.1.4	<p>-----NOTE----- Only required to be performed when stable thermal conditions are established at $\geq 90\%$ RTP. -----</p> <p>Verify RCS total flow rate is within the limit specified in the COLR by measurement.</p>	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 The RCS average temperature (T_{avg}) shall be ≥ 525 °F.

APPLICABILITY: MODE 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} not within limit.	A.1 Be in MODE 3.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS $T_{avg} \geq 525$ °F.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. -----NOTE----- Required Action D.2 shall be completed whenever this Condition is entered. ----- Requirements of LCO not met in other than MODE 1, 2, 3, or 4.	D.1 Initiate action to restore parameter(s) to within limit. <u>AND</u> D.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	<p>-----NOTE-----</p> <p>Only required to be performed during RCS heatup operations with fuel in the reactor vessel.</p> <p>-----</p> <p>Verify RCS pressure, RCS temperature, and RCS heatup rates are within the limits specified in Figure 3.4.3-1.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.4.3.2	<p>-----NOTE-----</p> <p>Only required to be performed during RCS cooldown operations with fuel in the reactor vessel.</p> <p>-----</p> <p>Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-2.</p>	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE	FREQUENCY
<p>SR 3.4.3.3 -----NOTE----- Only required to be performed during RCS heatup and cooldown operations with no fuel in the reactor vessel. -----</p> <p>Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-3.</p>	<p>In accordance with the Surveillance Frequency Control Program</p>
<p>SR 3.4.3.4 -----NOTE----- Only required to be performed during PHYSICS TESTS with RCS temperature ≤ 525 °F. -----</p> <p>Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.3-1.</p>	<p>In accordance with the Surveillance Frequency Control Program</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops – MODES 1 and 2

- LCO 3.4.4 Two RCS Loops shall be in operation, with:
- Four reactor coolant pumps (RCPs) operating; or
 - Three RCPs operating and THERMAL POWER restricted as specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCP not in operation in each loop.	A.1 Restore one non-operating RCP to operation.	18 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> LCO not met for reasons other than Condition A.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify required RCS loops are in operation.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two RCS loops inoperable. <u>OR</u> Required RCS loop not in operation.	C.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u> C.2 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.5.2	<p>-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. -----</p> <p>Verify correct breaker alignment and indicated power available to each required pump.</p>	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two required loops inoperable. <u>OR</u> Required loop not in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u> B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify required DHR or RCS loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.6.2	-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. ----- Verify correct breaker alignment and indicated power available to each required pump.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify required DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.2	Verify required SG secondary side water levels are ≥ 20 inches.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.3	<p>-----NOTE----- Not required to be performed until 24 hours after a required pump is not in operation. -----</p> <p>Verify correct breaker alignment and indicated power available to each required DHR pump.</p>	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No required DHR loop OPERABLE. <u>OR</u> Required DHR loop not in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u> B.2 Suspend all operations involving reduction in RCS water volume.	Immediately
	<u>AND</u> B.3 Initiate action to restore one DHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify required DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.2	<p>-----NOTE-----</p> <p>Not required to be performed until 24 hours after a required pump is not in operation.</p> <p>-----</p> <p>Verify correct breaker alignment and indicated power available to each required DHR pump.</p>	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level \leq 320 inches.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	Verify capacity of ES bus powered pressurizer heaters \geq 126 kW.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Verify pressurizer level does not represent a water solid condition.	30 minutes during RCS heatup and cooldown <u>AND</u> In accordance with the Surveillance Frequency Control Program
SR 3.4.11.2	Verify HPI is deactivated.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.3	Verify each pressurized CFT is isolated.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.4	<p>-----NOTE-----</p> <p>Verification of locked, sealed, or otherwise secured open vent path(s) only required to be performed every 31 days.</p> <p>-----</p> <p>Verify OPERABLE pressure relief capability.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.5	Perform CHANNEL CALIBRATION of ERV opening circuitry.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	<p>-----NOTE-----</p> <p>Only required to be performed in MODE 1 and 2, MODE 3 with RCS average temperature ≥ 500 °F.</p> <p>-----</p> <p>Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 2200 $\mu\text{Ci/gm}$.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 $\mu\text{Ci/gm}$.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Not required to be performed until 12 hours after establishment of steady state operation at or near operating pressure. 2. Not applicable to primary to secondary LEAKAGE. <p>-----</p> <p>Verify RCS operational LEAKAGE is within limits by performance of an RCS water inventory balance.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.4.13.2	<p>-----NOTE-----</p> <p>Not required to be performed until 12 hours after establishment of steady state operation.</p> <p>-----</p> <p>Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.</p>	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY										
SR 3.4.14.1	<p>-----NOTE----- Not required to be performed in MODES 3 and 4. -----</p> <p>Verify leakage from each RCS pressure isolation check valve, or pair of check valves, as applicable, is less than or equal to an equivalent of the Allowable Leakage Limit identified below at a differential test pressure ≥ 150 psid.</p> <table><thead><tr><th><u>Pressure Isolation Check Valve(s)</u></th><th><u>Allowable Leakage Limit</u></th></tr></thead><tbody><tr><td>DH-14A</td><td>≤ 5 gpm</td></tr><tr><td>DH-13A and DH-17</td><td>≤ 5 gpm total</td></tr><tr><td>DH-14B</td><td>≤ 5 gpm</td></tr><tr><td>DH-13B and DH-18</td><td>≤ 5 gpm total</td></tr></tbody></table>	<u>Pressure Isolation Check Valve(s)</u>	<u>Allowable Leakage Limit</u>	DH-14A	≤ 5 gpm	DH-13A and DH-17	≤ 5 gpm total	DH-14B	≤ 5 gpm	DH-13B and DH-18	≤ 5 gpm total	<p>In accordance with the INSERVICE TESTING PROGRAM</p> <p><u>AND</u></p> <p>Once prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months</p>
<u>Pressure Isolation Check Valve(s)</u>	<u>Allowable Leakage Limit</u>											
DH-14A	≤ 5 gpm											
DH-13A and DH-17	≤ 5 gpm total											
DH-14B	≤ 5 gpm											
DH-13B and DH-18	≤ 5 gpm total											
SR 3.4.14.2	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual high RCS pressure signal.	In accordance with the Surveillance Frequency Control Program										
SR 3.4.14.3	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual high RCS pressure signal: c. ≤ 340 psig for one valve; and d. ≤ 400 psig for the other valve.	In accordance with the Surveillance Frequency Control Program										
SR 3.4.14.4	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual Core Flood Tank isolation valve “not closed” signal.	In accordance with the Surveillance Frequency Control Program										
SR 3.4.14.5	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual Core Flood Tank isolation valve “not closed” signal.	In accordance with the Surveillance Frequency Control Program										

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of required reactor building atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of required reactor building atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.3	Perform CHANNEL CALIBRATION of required reactor building atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.4	Perform CHANNEL CALIBRATION of required reactor building sump monitor.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Core Flood Tanks (CFTs)

LCO 3.5.1 Two CFTs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > 800 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CFT inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One CFT inoperable for reasons other than Condition A.	B.1 Restore CFT to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Two CFTs inoperable.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce RCS pressure to ≤ 800 psig.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.1.1 Verify each CFT isolation valve is fully open.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.5.1.2	Verify borated water volume in each CFT is $\geq 970 \text{ ft}^3$ and $\leq 1110 \text{ ft}^3$.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify nitrogen cover pressure in each CFT is $\geq 560 \text{ psig}$ and $\leq 640 \text{ psig}$.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify boron concentration in each CFT is $\geq 2270 \text{ ppm}$.	<p>In accordance with the Surveillance Frequency Control Program</p> <p><u>AND</u></p> <p>-----NOTE----- Only required to be performed for affected CFT -----</p> <p>Once within 12 hours after each solution level increase of ≥ 0.2 feet that is not the result of addition from a borated water source of known concentration $\geq 2270 \text{ ppm}$</p>
SR 3.5.1.5	Verify power is removed from each CFT isolation valve operator.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) temperature > 350 °F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 Reduce RCS temperature to ≤ 350 °F.	12 hours
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.2.1 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.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.5.2.2	Verify each ECCS 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 3.5.2.3	Verify each ECCS 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.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.4	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.5	Verify, by visual inspection, each ECCS train reactor building sump suction inlet is not restricted by debris and screens show no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	<p>-----NOTE-----</p> <p>Only required to be performed when ambient air temperature is $< 40^{\circ}\text{F}$ or $> 110^{\circ}\text{F}$.</p> <p>-----</p> <p>Verify BWST borated water temperature is $\geq 40^{\circ}\text{F}$ and $\leq 110^{\circ}\text{F}$.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify BWST borated water level is ≥ 38.4 feet and ≤ 42 feet.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify BWST boron concentration is ≥ 2270 ppm and ≤ 2670 ppm.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	<p>-----NOTE-----</p> <ol style="list-style-type: none"> 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. <p>-----</p> <p>Perform required air lock leakage rate testing in accordance with the Reactor Building Leakage Rate Testing Program.</p>	In accordance with the Reactor Building Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each reactor building purge isolation valve is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	<p>-----NOTE-----</p> <p>Valves and blind flanges in high radiation areas may be verified by use of administrative means.</p> <p>-----</p> <p>Verify each reactor building isolation manual valve and blind flange that is located outside the reactor building and not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.3	<p>-----NOTE-----</p> <p>Valves and blind flanges in high radiation areas may be verified by use of administrative means.</p> <p>-----</p> <p>Verify each reactor building isolation manual valve and blind flange that is located inside the reactor building and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.</p>	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 reactor building isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.5	Verify each automatic reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.6 REACTOR BUILDING SYSTEMS

3.6.4 Reactor Building Pressure

LCO 3.6.4 Reactor building pressure shall be ≥ -1.0 psig and $\leq +3.0$ psig.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building pressure not within limits.	A.1 Restore reactor building pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 -----NOTE----- LCO 3.0.4.a is not applicable when entering Mode 4. ----- Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify reactor building pressure is ≥ -1.0 psig and $\leq +3.0$ psig.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify each reactor building spray manual, power operated, and automatic valve in each required flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.2	Operate each required reactor building cooling train fan unit for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.3	Verify each required reactor building cooling train cooling water flow rate is ≥ 1200 gpm.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.4	Verify each required reactor building 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 3.6.5.5	Verify each automatic reactor building spray valve in each required flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.6	Verify each required reactor building spray pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.7	Verify each required reactor building cooling train starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.8	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage

3.6 REACTOR BUILDING SYSTEMS

3.6.6 Spray Additive System

LCO 3.6.6 The Spray Additive System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.6.1 Verify each Spray Additive System 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.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.2 Verify sodium hydroxide tank solution volume is ≥ 9000 gallons.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.6.6.3	Verify sodium hydroxide tank solution concentration is > 6.0 wt% and < 8.5 wt.% NaOH.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.4	Verify each Spray Additive System automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	<p>-----NOTE-----</p> <p>Only required to be performed in MODES 1 and 2.</p> <p>-----</p> <p>Verify isolation time of each MSIV is within the limits specified in the INSERVICE TESTING PROGRAM.</p>	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.2.2	<p>-----NOTE-----</p> <ol style="list-style-type: none"> 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is < 750 psig. <p>-----</p> <p>Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.</p>	In accordance with the Surveillance Frequency Control Program

MFIVs, Main Feedwater Block Valves,
Low Load Feedwater Control Valves and
Startup Feedwater Control Valves
3.7.3

SURVEILLANCE		FREQUENCY
SR 3.7.3.2	-----NOTES-----	
	1. Only required to be performed in MODES 1 and 2.	
	2. Not required to be met when SG pressure is < 750 psig.	
	Verify that each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.4 Secondary Specific Activity

LCO 3.7.4 The specific activity of the secondary coolant shall be $\leq 0.1 \mu\text{Ci/gm}$ 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.	6 hours
	<u>AND</u> A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.4.1 Verify the specific activity of the secondary coolant is $\leq 0.1 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Be in MODE 4.	18 hours
D. Two EFW trains inoperable in MODE 1, 2, or 3.	D.1 -----NOTE----- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. ----- Initiate action to restore one EFW train to OPERABLE status.	Immediately
E. Required EFW train inoperable in MODE 4.	E.1 Initiate action to restore EFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	-----NOTE----- Not required to be performed for the turbine driven EFW pump, until 24 hours after reaching ≥ 750 psig in the steam generators. ----- Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE		FREQUENCY
SR 3.7.5.3	<p>-----NOTE----- Not required to be met in MODE 4 when steam generator is relied upon for heat removal. -----</p> <p>Verify each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.4	<p>-----NOTE----- Not required to be met in MODE 4 when steam generator is relied upon for heat removal. -----</p> <p>Verify each EFW pump starts automatically on an actual or simulated actuation signal.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.5	Verify proper alignment of the required EFW flow paths by verifying manual valve alignment from the "Q" condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever the unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days
SR 3.7.5.6	Verify that feedwater is delivered to each steam generator using the motor-driven EFW pump.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.6 Q Condensate Storage Tank (QCST)

LCO 3.7.6 The QCST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The QCST inoperable.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours <u>AND</u> Once per 12 hours thereafter
	<u>AND</u> A.2 Restore QCST to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4 without reliance on steam generator for heat removal.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 Verify QCST volume is $\geq 267,000$ gallons when required for both units and $\geq 107,000$ gallons when only required for Unit 1.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	<p>-----NOTE-----</p> <p>Isolation of SWS flow to individual components does not render the SWS inoperable.</p> <p>-----</p> <p>Verify each SWS 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.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.2	Verify each SWS 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.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.3	Verify each required SWS pump starts automatically on an actual or simulated signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.8 Emergency Cooling Pond (ECP)

LCO 3.7.8 The ECP shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Degradation of the ECP noted pursuant to SR 3.7.8.4 below or by other inspection.	A.1 Determine ECP remains acceptable for continued operation.	7 days
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> LCO not met for reasons other than Condition A.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.8.1 Verify that the indicated water level of the ECP is greater than or equal to that required for an ECP volume of 70 acre-ft.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.7.8.2	<p>-----NOTE-----</p> <p>Only required to be performed from June 1 through September 30.</p> <p>-----</p> <p>Verify average water temperature is ≤ 100 °F.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	<p>Perform soundings of the ECP to verify:</p> <ol style="list-style-type: none"> 1. A contained water volume of ECP ≥ 70 acre-feet, and 2. The minimum indicated water level needed to ensure a volume of 70 acre-feet is maintained. 	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.4	Perform visual inspection of the ECP to verify conformance with design requirements.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate each CREVS train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.9.3	Verify the CREVS automatically isolates the Control Room and switches into a recirculation mode of operation on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.4	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 3.7.10.1	Verify each CREACS train starts, operates for at least 1 hour, and maintains control room air temperature ≤ 84 °F D. B.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Verify system flow rate of 9900 cfm \pm 10%.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.11 Penetration Room Ventilation System (PRVS)

LCO 3.7.11 Two PRVS trains shall be OPERABLE.

-----NOTE-----
The penetration room negative pressure boundary may be opened intermittently under administrative controls.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PRVS train inoperable.	A.1 Restore PRVS train to OPERABLE status.	7 days
B. Two PRVS trains inoperable due to inoperable penetration room negative pressure boundary.	B.1 Restore penetration room negative pressure boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met. <u>OR</u> Both PRVS trains inoperable for reasons other than Condition B.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Operate each PRVS train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.11.2	Perform required PRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.11.3	Verify each PRVS train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool Water Level.

LCO 3.7.13 The spent fuel pool water level shall be \geq 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool water level not within limit.	A.1 -----NOTE----- LCO 3.0.3 is not applicable. ----- Suspend movement of irradiated fuel assemblies in the spent fuel pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify the spent fuel pool water level is \geq 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Boron Concentration

LCO 3.7.14 The spent fuel pool boron concentration shall be > 2000 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. 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 spent fuel pool.	Immediately
	<u>AND</u> A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the spent fuel pool boron concentration is > 2000 ppm.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. <u>AND</u>	6 hours
	F.2 -----NOTE----- LCO 3.0.4.a is not applicable when entering Mode 4. ----- Be in MODE 4.	12 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.2	-----NOTE----- All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. ----- Verify each DG starts from standby conditions and, in ≤ 15 seconds achieves "ready-to-load" conditions.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and follow, without shutdown, a successful performance of SR 3.8.1.2. <p>-----</p> <p>Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2475 kW and ≤ 2750 kW.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.4	Verify each day tank contains ≥ 160 gallons of fuel oil.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.5	Check for and remove accumulated water from each day tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.6	Verify the fuel oil transfer system operates to transfer fuel oil from storage tanks to the day tank.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.7</p> <p>-----NOTE----- This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. -----</p> <p>Verify automatic transfer of AC power sources to the selected offsite circuit and manual transfer to the alternate required offsite circuit.</p>	<p>In accordance with the Surveillance Frequency Control Program</p>
<p>SR 3.8.1.8</p> <p>-----NOTE----- All DG starts may be preceded by an engine prelube period. -----</p> <p>Verify on an actual or simulated loss of offsite power signal:</p> <ol style="list-style-type: none"> a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: <ol style="list-style-type: none"> 1. achieves “ready-to-load” conditions in ≤ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected shutdown load through automatic load sequencing timers, and 4. supplies connected loads for ≥ 5 minutes. 	<p>In accordance with the Surveillance Frequency Control Program</p>

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.9</p> <p>-----NOTE----- All DG starts may be preceded by an engine prelube period. -----</p> <p>Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal:</p> <ol style="list-style-type: none"> a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: <ol style="list-style-type: none"> 1. achieves “ready-to-load” conditions in ≤ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected emergency loads through load sequencing timers, and 4. supplies connected loads for ≥ 5 minutes. 	<p>In accordance with the Surveillance Frequency Control Program</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	<div>-----NOTES-----</div> <div>1. SR 3.8.1.3 is not required to be performed.</div> <div>2. The 15 second acceptance criteria of SR 3.8.1.2 is not applicable.</div> <div>-----</div> <div>For AC Sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources – Operating," except SR 3.8.1.4, SR 3.8.1.7, SR 3.8.1.8, and SR 3.8.1.9, are applicable.</div>	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. Required Action and associated Completion Time not met.</p> <p><u>OR</u></p> <p>One or more DGs with diesel fuel oil or required starting air subsystem not within limits for reasons other than Condition A, B, C, or D.</p>	E.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains $\geq 20,000$ gallons of fuel.	In accordance with the Surveillance Frequency Control Program
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
SR 3.8.3.3	Verify each DG required air start receiver pressure is ≥ 175 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.4	Check for and remove accumulated water from each fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.2	<p>Verify each battery charger supplies ≥ 300 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours.</p> <p><u>OR</u></p> <p>Verify each battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	<p>-----NOTE-----</p> <p>This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR.</p> <p>-----</p> <p>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 or a modified performance discharge test.</p>	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	<p>-----NOTE-----</p> <p>Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1.</p> <p>-----</p> <p>Verify each battery float current is ≤ 2 amps.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.2	Verify each battery pilot cell float voltage is ≥ 2.07 V.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	In accordance with the Surveillance Frequency Control Program
SR 3.8.6.5	Verify each battery connected cell float voltage is ≥ 2.07 V.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE	FREQUENCY
<p>SR 3.8.6.6 -----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. -----</p> <p>Verify battery capacity is $\geq 80\%$ of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.</p>	<p>In accordance with the Surveillance Frequency Control Program</p> <p><u>AND</u></p> <p>12 months when battery shows degradation, or has reached 85% of the expected life with capacity $< 100\%$ of manufacturer's rating</p> <p><u>AND</u></p> <p>24 months when battery has reached 85% of the expected life with capacity $\geq 100\%$ of manufacturer's rating</p>

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 -----NOTE----- LCO 3.0.4.a is not applicable when entering Mode 4. ----- Be in MODE 4.	12 hours
C. Two or more of the four inverters required by LCO 3.8.7.a and LCO 3.8.7.b inoperable.	C.1 Be in MODE 3. <u>AND</u>	12 hours
	C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to associated 120 VAC buses RS1, RS2, RS3, and RS4.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.5 Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by AC vital bus inverter(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.8.1 Verify correct inverter voltage and alignments to required 120 VAC vital buses.	In accordance with the Surveillance Frequency Control Program

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 to required AC, DC, and 120 VAC bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.2.4 Initiate actions to restore required AC, DC, and 120 VAC vital bus electrical power distribution subsystems to OPERABLE status.	Immediately
	<u>AND</u>	
	A.2.5 Declare associated required decay heat removal subsystem(s) inoperable.	Immediately
	<u>AND</u>	
	A.2.6 Enter applicable Conditions and Required Actions of LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," for LTOP features made inoperable by Electrical Power Distribution System.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments to required AC, DC, and 120 VAC vital bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

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

APPLICABILITY: MODE 6.

-----NOTE-----
Only applicable to the refueling canal when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	A.3 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.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.9.2.2	<p>-----NOTE-----</p> <p>Neutron detectors are excluded from CHANNEL CALIBRATION.</p> <p>-----</p> <p>Perform CHANNEL CALIBRATION.</p>	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required reactor building penetration is in the required status.	In accordance with the Surveillance Frequency Control Program
SR 3.9.3.2	<p>-----NOTE-----</p> <p>Not required to be met for reactor building isolation valves and reactor building purge isolation valves in penetrations closed to comply with LCO c.1.</p> <p>-----</p> <p>Verify each required reactor building isolation valve and each reactor building purge isolation valve actuates to the isolation position.</p>	In accordance with the Surveillance Frequency Control Program
SR 3.9.3.3	Perform CHANNEL CALIBRATION of reactor building purge exhaust radiation monitor.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No DHR loop OPERABLE or in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	<u>AND</u>	
	B.2 Initiate action to restore one DHR loop to OPERABLE status and to operation.	Immediately
	<u>AND</u>	
	B.3 Close all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one DHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to each required DHR pump.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.6 Refueling Canal Water Level

LCO 3.9.6 Refueling canal water level shall be maintained \geq 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of irradiated fuel assemblies within the reactor building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within the reactor building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling canal water level is \geq 23 feet above the top of irradiated fuel assemblies seated within the reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The program shall include the following:

- a. Preventive maintenance and periodic visual inspection requirements; and
- b. Integrated leak test requirements for each system at a frequency in accordance with the Surveillance Frequency Control Program. The provisions of SR 3.0.2 are applicable.

5.5.3 Iodine Monitoring

This program provides controls that ensure the capability to accurately determine the airborne iodine concentration under accident conditions. The program shall include the following:

- a. Training of personnel;
- b. Procedures for monitoring; and
- c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;
- b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming 10 CFR 20, Appendix B, Table II, Column 2;

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.5 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

- a. The definition of the CRE and the CRE boundary.
- b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.
- c. Requirements for (i) determining the unfiltered air leakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," 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. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency in accordance with the Surveillance Frequency Control Program. The results shall be trended and used as part of the CRE boundary assessment specified in TS 5.5.5.c.
- e. The quantitative limits on unfiltered air leakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air leakage measured by the testing described in paragraph c. The unfiltered air leakage limit for radiological challenges is the leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air leakage 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 leakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel. Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program inspection frequencies.

5.5.8 Surveillance Frequency Control Program

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

- a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
- b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
- c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
 1. an API gravity or an absolute specific gravity within limits,
 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
 3. water and sediment within limits;
- b. Within 31 days following addition of new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil;
- c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested based on ASTM D-2276, Method A-2 or A-3 at a Frequency in accordance with the Surveillance Frequency Control Program; and
- d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance Frequencies.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
 1. A change in the TS incorporated in the license; or
 2. A change to the updated SAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

Proposed changes that do meet these criteria shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the SAR.

ATTACHMENT 5

1CAN031801

**PROPOSED TECHNICAL SPECIFICATION BASES CHANGES
(Information only)**

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.1.1 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~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 may include performing a boron concentration analysis, and complete the calculation.~~

REFERENCES

1. SAR, Section 1.4, GDC 26.
 2. SAR, Chapter 3.
 3. 10 CFR 50.36.
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ACTIONS (continued)

A.1 and A.2 (continued)

If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the appropriate reactivity parameter may be renormalized, and operation in MODE 1 may continue. If operational restrictions or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing operating restrictions or surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity balance cannot be restored to within the $\pm 1\%$ $\Delta k/k$ limit, the unit must be brought to a MODE in which the LCO does not apply. As a conservative measure, the unit must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by Required Action A.1 of LCO 3.1.1 would occur. The allowed Completion Time of 6 hours is reasonable, based on operating experience to reach the required unit conditions from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by a periodic reactivity balance calculation that compares the predicted core reactivity to the actual core reactivity condition (net reactivity of zero condition). The comparison is made considering that core conditions are fixed or stable, including CONTROL ROD and APSR positions, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed once prior to entering MODE 1 after each fuel loading as an initial check on core reactivity conditions and design calculations at BOC. A Note is included in the SR to indicate that the normalization of predicted core reactivity to the measured value may take place within the first 60 effective full power days (EFPD) after each fuel loading. ~~The required Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1 is acceptable, based on the slow rate of core reactivity changes due to fuel depletion and the presence of other indicators (QPT, etc.) for prompt indication of an anomaly.~~ The 60 EFPD after entering MODE 1 allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.

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

Another Note is included in the SRs to indicate that the performance of the Surveillance is not required for entry into MODE 2.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual CONTROL RODS are aligned within 6.5% of their group average height limits ~~at a 12-hour Frequency~~ allows the operator to detect a rod that is beginning to deviate from its expected position. The ~~Surveillance specified Frequency is controlled under the Surveillance Frequency Control Program~~ takes into account other CONTROL ROD position information that is continuously available to the operator in the control room, so that during actual CONTROL ROD motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each CONTROL ROD is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each CONTROL ROD could result in radial tilts. Exercising each individual CONTROL ROD ~~every 92 days~~ provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each CONTROL ROD by approximately 1.5% (approximately 2 inches) will not cause radial or axial power tilts, or oscillations, to occur. No additional allowances for instrument uncertainty are required to be incorporated in the implementing procedures for this parameter. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program~~ The 92-day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between typical performances of SR 3.1.4.2 (determination of CONTROL ROD OPERABILITY by movement), if a CONTROL ROD(S) is discovered to be immovable, but is otherwise determined to be capable of being fully inserted, the CONTROL ROD(S) may continue to be considered OPERABLE unless inoperable for some other reason. At any time, if a CONTROL ROD(S) is immovable, a determination of the capability to fully insert (OPERABILITY) the CONTROL ROD(S) must be made and appropriate action taken. Provided the CONTROL ROD(S) are capable of full insertion, no further assessment is necessary.

When one or more CONTROL ROD(S) have failed to fully insert, additional assessment of CONTROL ROD OPERABILITY is required. Any CONTROL ROD that fails to fully insert may be considered OPERABLE provided:

- a. The CONTROL ROD is $\geq 90\%$ inserted or, during power operation, a determination has been made that the CONTROL ROD will insert to at least 90% upon a reactor trip.
- b. SDM is verified. The SDM shall assume the affected CONTROL ROD remains 10% withdrawn. This SDM correction for the affected CONTROL ROD(S) shall be included in appropriate reactivity control procedures for the remaining reactor operational cycle or until the condition causing the failure to fully insert is eliminated.
- c. The cause or probable cause of the condition, if left uncorrected, provides reasonable assurance that the CONTROL ROD will insert to $\geq 90\%$ inserted upon future reactor trip.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that each safety rod is fully withdrawn ensures the safety rods are available to provide reactor shutdown capability.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~Verification that individual safety rod positions are fully withdrawn at a 12-hour Frequency allows the operator to detect a safety rod beginning to deviate from its expected position. Also, the 12-hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of the safety rods.~~

REFERENCES

1. SAR, Section 1.4, GDC 10, GDC 26, and GDC 28.
 2. 10 CFR 50.46.
 3. SAR, Chapters 3 and 4.
 4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2.
 5. 10 CFR 50.36.
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ACTIONS (continued)

A.1 (continued)

An alternate to realigning a single misaligned APSR to the group average position is to align the remainder of the APSR group to the position of the misaligned or inoperable APSR, while maintaining APSR insertion, in accordance with the limits in the COLR. This restores the alignment requirements. Deviations up to 2 hours will not cause significant xenon redistribution to occur. This alternative assumes the APSR group movement does not cause the limits of LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," to be exceeded. For this reason, APSR group movement is only practical for instances where small movements of the APSR group are sufficient to re-establish APSR alignment.

The reactor may continue in operation with the APSR misaligned if the limits on power peaking are surveilled within 2 hours to determine if power peaking is still within limits. Also, since any additional movement of the APSRs may result in additional imbalance, Required Action A.1 also requires the power peaking surveillance to be performed again within 2 hours after each APSR movement.

B.1

The unit must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems. In MODE 3, APSR group alignment limits are not required because the reactor is not generating significant THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification ~~at a 12 hour Frequency~~ that individual APSR positions are within 6.5% of the group average height limits allows the operator to detect an APSR beginning to deviate from its expected position. In addition, APSR position is continuously available to the operator in the control room so that during actual APSR motion, deviations can immediately be detected. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

REFERENCES

1. SAR, Section 1.4, GDC 10 and GDC 28.
2. 10 CFR 50.46.
3. 10 CFR 50.36.

APPLICABILITY

In MODES 1 and 2, OPERABILITY of the position indicator channel is required, since the reactor is, or is capable of, generating THERMAL POWER in these MODES. In MODES 3, 4, 5, and 6, Applicability is not required because the reactor is shut down with the required minimum SDM and is not generating significant THERMAL POWER.

ACTIONS

A.1

If the required position indicator channel is inoperable for one or more rods, the position of the CONTROL ROD or APSR is not known with certainty. Therefore, each affected CONTROL ROD or APSR must be declared inoperable, and the limits of LCO 3.1.4 or LCO 3.1.6 apply. The required Completion Time for declaring the rod(s) inoperable is immediately. Therefore LCO 3.1.4 or LCO 3.1.6 is entered immediately, and the required Completion Times for the appropriate Required Actions in those LCOs apply without delay.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

A CHANNEL CHECK of the required position indication channel ensures that position indication for each CONTROL ROD and APSR remains OPERABLE and accurate. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. However, this CHANNEL CHECK will be used to detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

When compared to other channels, the agreement criteria between channels is determined by the unit staff. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required position indicator channel.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The required Frequency of 12 hours is adequate for verifying that no degradation in system OPERABILITY has occurred.~~

SR 3.1.7.2

A CHANNEL CALIBRATION of the required position indication channel verifies that the channel responds within the necessary range and accuracy.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency of 18 months is based on operating experience and consistency with the typical industry refueling cycle.~~

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that THERMAL POWER is $\leq 85\%$ RTP ensures that the required additional thermal margin has been established prior to and during PHYSICS TESTS. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The required Frequency of once per hour allows the operator adequate time to determine any degradation of the established thermal margin during PHYSICS TESTS.~~

SR 3.1.8.2

Verification that core LHRs are within their limits ensures that core LHR and departure from nucleate boiling ratio will remain within their limits, while one or more of the LCOs that normally control these design limits are out of specification. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The required Frequency of 2 hours allows the operator adequate time for collecting a flux map and for performing the LHR verification, based on operating experience.~~ If SR 3.2.5.1 is not met, PHYSICS TESTS are suspended and LCO 3.2.5 applies. This Frequency is more conservative than the Completion Time for restoration of the individual LCOs that preserve the LHR limits.

This SR is modified by a Note that requires performance only when THERMAL POWER is greater than 20% RTP. This establishes a performance requirement that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

SR 3.1.8.3

Verification that the nuclear overpower trip setpoint is within the limit specified for each PHYSICS TEST ensures that core protection at the reduced power level is established during the PHYSICS TESTS. Performing the verification once within 8 hours prior to the performance of PHYSICS TESTS at each testing plateau allows the operator adequate time for verifying the established trip setpoint before initiating PHYSICS TESTS.

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

- a. Reactor Coolant System (RCS) boron concentration;
- b. CONTROL ROD position;
- c. Doppler defect;
- d. Fuel burnup based on gross thermal energy generation;
- e. Samarium concentration;
- f. Xenon concentration; and
- g. Moderator defect.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
 2. 10 CFR 50.59.
 3. SAR, Section 3A.9.
 4. SAR, Section 13.3, 13.4 and 13.6.
 5. SAR, Section 13.4, Table 13-2.
 6. 10 CFR 50.36.
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ACTIONS (continued)

B.1 and B.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. A Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs.

C.1

If the nuclear overpower trip setpoint is $> 5\%$ RTP, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

If the nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit function is inoperable, then 1 hour is allowed for the operator to restore the functions to OPERABLE status or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

The nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit function is not required when the reactor power level is above the operating range of the instrumentation channel. For example, if the reactor power level is above the source range channel operating range, then only the intermediate range high startup rate CONTROL ROD withdrawal inhibit is required to be functional.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that THERMAL POWER is $\leq 5\%$ RTP ensures that local LHR, DNBR, and RCS pressure limits are not violated and that entry into Actions Condition A is performed promptly. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~Hourly verification is adequate for the operator to determine any change in core conditions, such as xenon redistribution occurring after a THERMAL POWER reduction, that could cause THERMAL POWER to exceed the specified limit.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.9.2

Verification that the nuclear overpower trip setpoint is within the limit specified for PHYSICS TESTS ensures that core protection at the reduced power level is established during PHYSICS TESTS. Performing the verification once within 8 hours prior to the performance of PHYSICS TESTS allows the operator adequate time for verifying the established trip setpoint before initiating PHYSICS TESTS.

SR 3.1.9.3

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

- a. RCS boron concentration;
- b. CONTROL ROD position;
- c. RCS average temperature;
- d. Fuel burnup based on gross thermal energy generation;
- e. Samarium concentration;
- f. Xenon concentration;
- g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH);
- h. Moderator defect, when above the POAH; and
- i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the POAH, and the fuel temperature will be changing at the same rate as the RCS.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

ACTIONS (continued)

D.2.2

The SDM and ejected rod worth limit can also be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion setpoints in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the unit systems. Operation for up to 2 hours in the restricted operation region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions or configurations and limits the potential for an adverse xenon redistribution.

E.1

If the Required Actions and associated Completion Times of Conditions C or D are not met, then the reactor is placed in MODE 3, in which this LCO does not apply. This Action ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the amount of time required to reach MODE 3 from RTP without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This Surveillance ensures that the sequence and overlap limits are not violated. ~~TheA Surveillance Frequency is controlled under the Surveillance Frequency Control Program of 12 hours is acceptable because little rod motion occurs during this period due to fuel burnup. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.~~

SR 3.2.1.2

Verification of the regulating rod insertion setpoints as specified in the COLR ~~at a Frequency of 12 hours~~ is sufficient to detect regulating rod banks that may be approaching the group insertion setpoints, ~~because little rod motion due to fuel burnup occurs in 12 hours.~~ The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.~~

SR 3.2.1.3

Prior to achieving criticality, an estimated critical position for the CONTROL RODS is determined. Verification that SDM meets the minimum requirements ensures that sufficient SDM capability exists with the CONTROL RODS at the estimated critical position if it is

ACTIONS (continued)

B.1

If the Required Action and associated Completion Time are not met, the reactor must be placed in MODE 3, in which this LCO does not apply. This action ensures that the fuel does not continue to be depleted in an unintended burnup distribution. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 3 from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

Fuel cycle designs that allow APSR withdrawal near end of cycle (EOC) only permit reinsertion of APSRs at $\leq 30\%$ power during plant shutdown for the refueling outage. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Verification that the APSRs are within their insertion setpoints at a 12-hour Frequency is sufficient to ensure that the APSR insertion setpoints are preserved. The 12-hour Frequency required for performing this verification is sufficient because APSRs are positioned by manual control and are normally moved infrequently. The Frequency takes into account other information available in the control room for monitoring the axial power distribution in the reactor core.~~

REFERENCES

1. SAR, Section 1.4, GDC 10 and GDC 26.
 2. 10 CFR 50.46.
 3. SAR, Chapter 14.
 4. 10 CFR 50.36.
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SURVEILLANCE REQUIREMENTS (continued)

Figure B 3.2.3-1 (Minimum Incore Detector System for AXIAL POWER IMBALANCE Measurement) depicts an example of this configuration. This arrangement is chosen to reduce the uncertainty in the measurement of the AXIAL POWER IMBALANCE by the Minimum Incore Detector System. For example, the requirement for placing one detector of each of the three strings at the core midplane puts three detectors in the central region of the core where the neutron flux tends to be higher. It also helps prevent measuring an AXIAL POWER IMBALANCE that is excessively large when the reactor is operating at low THERMAL POWER levels. The third requirement for placement of detectors (i.e., radial asymmetry) reduces uncertainty by measuring the neutron flux at core locations that are not radially symmetric.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Verification of the AXIAL POWER IMBALANCE indication every 12 hours ensures that the AXIAL POWER IMBALANCE setpoints are not violated and takes into account other information and alarms available in the control room. This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or control rod drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, can be discovered by the operator before the specified limits are violated.

REFERENCES

1. 10 CFR 50.46.
 2. 10 CFR 50.36.
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SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

Checking the QPT indication every 7 days ensures that the operator can determine whether the plant computer software and Incore Detector System inputs for monitoring QPT are functioning properly, and takes into account other information and alarms available to the operator in the control room. This procedure allows the QPT mechanisms, such as xenon redistribution, burnup gradients, and CONTROL ROD drive mechanism malfunctions, which can cause slow development of a QPT, to be detected. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~Operating experience has confirmed the acceptability of a Surveillance Frequency of 7 days.~~

Following restoration of the QPT to within the setpoint, operation at $\geq 95\%$ RTP may proceed provided the QPT is determined to remain within the setpoint at the increased THERMAL POWER level. In case QPT exceeds the setpoint for more than 24 hours (Condition A), the potential for xenon redistribution is greater. Therefore, the QPT is monitored for 12 consecutive hourly intervals to determine whether the period of any oscillation due to xenon redistribution causes the QPT to exceed the setpoint again.

REFERENCES

1. 10 CFR 50.46
 2. BAW 10122A, "Normal Operating Controls," Rev. 1, May 1984.
 3. 10 CFR 50.36
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ACTIONS (continued)

F.1

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition F, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 45% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 45% RTP from full power conditions in an orderly manner without challenging unit systems.

G.1

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition G, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 10% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 10% RTP from full power conditions in an orderly manner without challenging unit systems.

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION testing.

The SRs are modified by a Note which directs the reader to Table 3.3.1-1 to determine the correct SRs to perform for each RPS Function.

SR 3.3.1.1

Performance of the CHANNEL CHECK ~~once every 12 hours~~ provides reasonable assurance of prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same 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 two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of factors including channel instrument uncertainties. 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. 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 is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are, where practical, verified to be reading at the bottom of the range and not failed downscale.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1 (continued)

The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. For example, during a power increase near the top of the scale for the intermediate range monitors, a power range monitor reading is expected with at least one decade overlap. Without such an overlap, the power range monitors are considered inoperable unless it is clear that an intermediate range monitor inoperability is responsible for the lack of the expected overlap.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency is based on operating experience that demonstrates 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 protective function due to failure of redundant channels.~~ The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

For Functions that trip on a combination of several measurements, such as the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE Function, the CHANNEL CHECK must be performed on each input.

SR 3.3.1.2

This SR is the performance of a heat balance calibration for the power range channels ~~periodically every 96 hours~~ and once within 24 hours after a THERMAL POWER change of $\geq 10\%$ RTP in one direction, when reactor power is $\geq 20\%$ RTP. The heat balance calibration consists of a comparison of the results of the calorimetric with the power range channel output. The outputs of the power range channels are calibrated to the calorimetric. Note 1 to the SR states if the absolute difference between the calorimetric and the Nuclear Instrumentation System (NIS) channel is $> 2\%$ RTP, the NIS channel is not declared inoperable but must be adjusted. If the NIS channel cannot be properly adjusted, the channel is declared inoperable. Note 2 clarifies that this Surveillance is required only if reactor power is $\geq 20\%$ RTP and that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP.

Two calorimetric calculations are routinely performed. One relies upon primary system parameters and the other relies upon secondary system parameters. The primary calorimetric is generally less accurate than the secondary calorimetric at higher power levels and more accurate at lower power levels. For comparison to the nuclear instrumentation, between 0 and 15% power, only the primary calorimetric (heat balance) is considered. From 15 to 100% power the calorimetric is weighted linearly with only the secondary heat balance being considered at 100% power.

The power range channel's output shall be adjusted consistent with the calorimetric results if the absolute difference between the calorimetric and the power range channel's output is $> 2\%$ RTP. The value of 2% is adequate because this value is assumed in the safety analyses of SAR, Chapter 14 (Ref. 2). These checks and, if necessary, the adjustment of the power range channels ensure that channel accuracy is maintained within the analyzed error margins.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 96-hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds 2% in any 96-hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

A comparison of power range nuclear instrumentation channels against incore detectors shall be performed ~~periodically at a 31-day Frequency~~ when reactor power is $\geq 20\%$ RTP. The SR is modified by two Notes. Note 2 clarifies that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP. Note 1 states if the absolute difference between the power range and incore AXIAL POWER IMBALANCE measurements is $\geq 2\%$ RTP, the power range channel is not inoperable, but an adjustment of the measured imbalance to agree with the incore measurements is necessary. If the power range channel cannot be properly recalibrated, the channel is declared inoperable. The calculation of the Allowable Value envelope assumes a difference in out of core to incore AXIAL POWER IMBALANCE measurements of 2.5%. Additional inaccuracies beyond those that are measured are also included in the setpoint envelope calculation. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 31-day Frequency is adequate, considering that long term drift of the excore linear amplifiers is small and burnup of the detectors is slow. Also, 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. The slow changes in neutron flux during the fuel cycle can also be detected at this interval.~~

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed to ensure that the entire channel will perform the intended function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 31 days is based on operating experience, which has demonstrated through high reliability of the instrumentation, that failure of more than one channel, of a given Function, in any 31-day interval is rare.~~ Testing in accordance with this SR is normally performed on a rotational basis, with one channel being tested each week. Testing one channel each week reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant channel. The automatic bypass removal feature is verified for the turbine oil pressure trip and the main feedwater pump oil pressure trip functions during the CHANNEL FUNCTIONAL TEST.

SR 3.3.1.5

A Note to the Surveillance indicates that neutron detectors are excluded from CHANNEL CALIBRATION. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between

successive tests. CHANNEL CALIBRATION shall find that instrument errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATION must be performed consistent with the assumptions of the setpoint analysis. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature (RTD) sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is justified by the assumption of at least an 18 month calibration interval in the determination of the allowable magnitude of equipment drift in the setpoint analysis.

ACTIONS (continued)

A.1.1, A.1.2, and A.2 (continued)

To ensure the trip signal is registered in the other channels, Required Action A.2 requires that the inoperable RTM be removed from the cabinet. This action causes the electrical interlocks to indicate a tripped channel in the remaining three RTMs. Operation in this condition is allowed indefinitely because the actions put the RPS into a one-out-of-three configuration. The 1 hour Completion Time is sufficient time to perform the Required Actions.

B.1, B.2.1, and B.2.2

Condition B applies if two or more RTMs are inoperable in MODE 1, 2, or 3, or if the Required Actions and associated Completion Time of Condition A are not met in MODE 1, 2, or 3. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 with all CRD trip breakers open or with power to all CRD trip breakers removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

Condition C applies if two or more RTMs are inoperable in MODE 4 or 5, or if the Required Actions and associated Completion Times are not met in MODE 4 or 5. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by opening all CRD trip breakers or removing power from all CRD trip breakers. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove all power to the CRD System without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

The SRs include performance of a CHANNEL FUNCTIONAL TEST ~~every 92 days~~. This test shall verify the OPERABILITY of the RTM and its ability to receive and properly respond to channel trip and reactor trip signals.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program ~~The Frequency of 92 days is based on operating experience, which has demonstrated through high reliability of the instrumentation, that failure of more than one RTM in any 92 day interval is rare (Ref. 3).~~

~~Testing in accordance with this SR is normally performed on a rotational basis, with one RTM being tested each 23 days. Testing one RTM each 23 days reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant RTM.~~

ACTIONS (continued)

E.1 and E.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met in MODE 4 or 5, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, all CRD trip breakers must be opened or power to all CRD trip breakers removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove power from all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 is to perform a CHANNEL FUNCTIONAL TEST ~~every 92 days~~. This test verifies the OPERABILITY of the trip devices by actuation of the end devices. Also, this test independently verifies the undervoltage and shunt trip mechanisms of the trip breakers. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program—The Frequency of 92 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 92 day interval is a rare event (Ref. 3).~~

~~Testing in accordance with this SR is normally performed on a rotational basis with one channel being tested each 23 days. Testing one channel each 23 days reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant trip device.~~

REFERENCES

1. SAR, Chapter 7.
 2. 10 CFR 50.36.
 - ~~3. BAW 10167A, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," Supplement 3, "Justification for Increasing the Trip Device Test Interval," February 1998.~~
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SR 3.3.5.1

Performance of the CHANNEL CHECK ~~every 12 hours~~ provides reasonable assurance for prompt identification of a gross failure of instrumentation. 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 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 drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between CHANNEL CALIBRATIONS.

Agreement criteria are determined by the unit staff, based on a combination of factors including channel instrument uncertainties. 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program ~~The Frequency is based on operating experience that demonstrates 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 protective function due to failure of redundant channels.~~ The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

SR 3.3.5.2

A CHANNEL FUNCTIONAL TEST is performed on each required ESAS analog instrument channel to ensure the entire channel will perform the intended functions. Any setpoint adjustment shall be consistent with the assumptions of the setpoint calculations.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program ~~The Frequency of 31 days is based on unit operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31-day interval is a rare event.~~ The RCS low pressure automatic bypass removal feature is verified during its CHANNEL FUNCTIONAL TEST.

SR 3.3.5.3

CHANNEL CALIBRATION is a complete check of the analog instrument channel, including the sensor. The 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 to ensure that the analog instrument channel remains OPERABLE between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint calculations. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint calculations.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program ~~This Frequency is justified by the assumption of at least an 18-month calibration interval to determine the magnitude of equipment drift in the setpoint calculations.~~

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the ESAS manual initiation. This test verifies that the initiating circuitry is OPERABLE and will actuate the digital actuation logic channels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The 18-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency is demonstrated to be sufficient, based on operating experience, which shows these components usually pass the Surveillance when performed on the 18-month Frequency.~~

REFERENCES

1. 10 CFR 50.36.
 2. BAW-2441-A, Revision 2, Risk Informed Justification for LCO End-State Changes, September 2006.
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SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

SR 3.3.7.1 is the performance of a CHANNEL FUNCTIONAL TEST ~~on a 31 day Frequency. The test demonstrates that each digital actuation logic channel successfully performs the two out of three logic combinations every 31 days.~~ The test simulates the required one-out-of-three inputs to the logic circuit and verifies the successful operation of the digital actuation logic. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval.~~ The CHANNEL FUNCTIONAL TEST performed for the Reactor Building Spray System Logic Channels shall include testing of the associated spray pump, spray valves, and chemical additive valve logic channels.

REFERENCES

1. 10 CFR 50.46.
 2. BAW-10192PA, "BWNT Loss-of-Coolant Accident Evaluation Model for OTSG Plants, Volumes I and II," June 1998.
 3. 10 CFR 50.36.
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B.1

Condition B applies if the Required Action and associated Completion Time of Condition A are not met.

Required Action B.1 ensures that Required Actions for affected diesel generator inoperabilities are initiated. Depending on the DG(s) affected, the appropriate Actions specified in LCO 3.8.1, "AC Sources - Operating," are required immediately.

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

Performance of the CHANNEL CHECK ~~once every 7 days~~ provides reasonable assurance for prompt identification of a gross failure of instrumentation. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on operating experience that demonstrates channel failure is rare. Since the probability of random failure in any 7 day period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of this instrumentation.~~

SR 3.3.8.2

The Note allows channel bypass for testing of the loss of voltage Function without entering the associated Conditions and Required Actions, although during this time period it cannot actuate a diesel start. This allowance is based on the assumption that 4 hours is the average time required to perform channel Surveillance. The 4 hour testing allowance does not significantly reduce the probability that the DG will start when necessary. It is not acceptable to remove channels from service for more than 4 hours to perform required Surveillance testing without declaring the channel inoperable.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The setpoints and the response to a loss of voltage and a degraded voltage test shall include a single point verification that the trip occurs within the required delay time. CHANNEL CALIBRATION shall verify that setpoints are within the required ranges. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on the reliability of the components, on operating experience which demonstrates channel failure is rare, and on consistency with the typical industry refueling cycle, and is justified by the assumption of at least an 18 month calibration interval in the determination of equipment drift.~~

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance of prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same 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 drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that 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 is required, the CHANNEL CHECK will only verify that they are off scale in the same direction.

The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. For example, during a power reduction near the bottom of the scale for the intermediate range monitors, a source range monitor reading is expected with at least one decade overlap. Without such an overlap, the source range monitors are considered inoperable unless it is clear that an intermediate range monitor inoperability is responsible for the lack of the expected overlap.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency is based on operating experience that demonstrates 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 protective function due to failure of redundant channels.~~ The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channel. When operating with only one channel OPERABLE, CHANNEL CHECK is still required. However, in this condition, a redundant source range may not be available for comparison. CHANNEL CHECK may still be performed via comparison with intermediate range detectors, if available, and verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

SR 3.3.9.2

For a source range neutron flux channel, CHANNEL CALIBRATION is a complete check and readjustment of the channel from the preamplifier input to the indicator. This test verifies the channel responds to measured parameters within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel at a setpoint which accounts for instrument drift to ensure that the instrument channel remains operational between successive tests.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.2 (continued)

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult, and there is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices with minimal drift. Finally, the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency of 18 months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an 18 month interval, such that the instrument is not adversely affected by drift.~~

REFERENCES

1. 10 CFR 50.36.
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APPLICABILITY (continued)

The intermediate range instrumentation is designed to detect power changes when the power range and source range instrumentation cannot provide reliable indications, e.g., during initial criticality and power escalation. Since those conditions can exist in, or propagate from, all of these MODES, the intermediate range instrumentation must be OPERABLE.

ACTIONS

A.1 and A.2

With the required intermediate range neutron flux channel inoperable when THERMAL POWER is $\leq 5\%$ RTP, the operators must place the reactor in the next lowest condition for which the intermediate range instrumentation is not required. This involves providing power level indication on the source range instrumentation by immediately suspending operations involving positive reactivity changes and, within 1 hour, placing the reactor in the tripped condition with the CRD trip breakers open. RCS temperature changes are permitted provided the effects of such changes are accounted for in the SDM calculations. The Completion Times are based on unit operating experience and allow the operators sufficient time to manually insert the CONTROL RODS prior to opening the CRD breakers.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

Performance of the CHANNEL CHECK ~~once every 12 hours~~ provides reasonable assurance of prompt identification that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same 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 drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that 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. Off scale low current loop channels are verified, where practical to be reading at the bottom of the range and not failed low.

The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. For example, during a power increase near the top of the scale for the source range monitors, an intermediate range monitor reading is expected with at least one decade overlap. Without such an overlap, the intermediate range monitors are considered inoperable unless it is clear that a source range monitor inoperability is responsible for the lack of the expected overlap.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.10.1 (continued)

Further, during a power reduction near the bottom of the scale for the power range monitors, an intermediate range monitor reading is expected with at least one decade overlap. Without such an overlap, the intermediate range monitors are considered inoperable unless it is clear that a power range monitor inoperability is responsible for the lack of the expected overlap.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency, about once every shift, is based on operating experience that demonstrates 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 protective function due to failure of redundant channels.~~ The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channel.

When operating with only one channel OPERABLE, CHANNEL CHECK is still required. However, in this condition, a redundant intermediate range is not available for comparison. CHANNEL CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE intermediate range channel is energized and indicates a value consistent with current unit status.

SR 3.3.10.2

A CHANNEL FUNCTIONAL TEST, of the required intermediate range instrument channel, verifies proper operation of the channel each 31 days. Monthly testing provides reasonable assurance that the instrument channel will function, if required, to provide indication during MODE 2 and during unanticipated reactivity excursions from MODES 3, 4, or 5. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

SR 3.3.10.3

For intermediate range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels, from the preamplifier input to the indicators. This test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel at a setpoint which accounts for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by demonstrated instrument reliability over an 18-month interval such that the instrument is not adversely affected by drift.~~

SURVEILLANCE REQUIREMENTS

A Note indicates that the SRs for each EFIC instrumentation Function are identified in the SRs column of Table 3.3.11-1. Individual EFIC subgroup relays must also be tested, one at a time, to verify the individual EFIC components will actuate when required. Some components cannot be tested at power since their actuation might lead to unit trip or equipment damage. These are specifically identified and must be tested when shut down. The various SRs account for individual functional differences and for test frequencies applicable specifically to the Functions listed in Table 3.3.11-1. The operational bypasses associated with each EFIC instrumentation channel are also subject to these SRs to ensure OPERABILITY of the EFIC instrumentation channel.

SR 3.3.11.1

Performance of the CHANNEL CHECK ~~once every 12 hours~~ provides reasonable assurance for prompt identification of a gross failure of instrumentation. 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 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 drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of factors including channel instrument uncertainties. 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. 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 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, where practical, to be reading at the bottom of the range and not failed downscale.

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on operating experience that demonstrates 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 protective function due to failure of redundant channels.~~ 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.

SR 3.3.11.2

A CHANNEL FUNCTIONAL TEST verifies the function of the automatic bypass removal feature, required trip, interlock, and alarm functions of the channel. Setpoints for trip functions must be found within the Allowable Value. (Note that the values for the bypass removal functions are identified in the Applicable MODES or Other Specified Condition column of Table 3.3.11-1 as limits on applicability for the trip Functions.) Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.11.2 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency of 31 days is based on unit operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.~~

This SR is modified by two notes. For the SG Level – Low function, if the as-found trip setpoint is found to be non-conservative with respect to the Allowable Value specified in TSs, the channel is declared inoperable and the associated TS action statement must be followed. If the as-found trip setpoint is found to be conservative with respect to the Allowable Value and outside the as-found predefined acceptance criteria band of ± 1.08 inches from the previous as-left value, but is determined to be functioning as required and can be reset to a value equal to the Limiting Trip Setpoint or a value more conservative than the Limiting Trip Setpoint, then the channel may be considered to be operable. If it cannot be determined that the instrument channel is functioning as required, the channel is declared inoperable and the associated TS actions must be followed. If the as-found trip setpoint is outside the as-found predefined acceptance criteria band, the condition must be entered into the corrective action program for further evaluation. The notes for the Channel Functional Test do not apply to the verification of the time delay.

SR 3.3.11.3

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channels adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis. The notes contained in SR 3.3.11.2 are also applicable to the CHANNEL CALIBRATION.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency is based on the assumption of at least an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.~~

REFERENCES

1. 10 CFR 50.62.
2. SAR, Chapter 7.
3. SAR, Chapter 14.
4. Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001.

ACTIONS (continued)

C.1

With one or both required manual initiation switches of one or more EFIC Function(s) inoperable in both actuation trains, one actuation train for each Function must be restored to OPERABLE status within 1 hour. With the train restored, the second train must be placed in the appropriate condition within 72 hours per Required Action A.1 or B.1, as applicable. Compliance with these actions ensures the single-failure criterion is met. The Completion Time allotted to restore the train allows the operator to take all the appropriate actions for the failed train and still ensures that the risk involved in operating with the failed train is acceptable.

D.1 and D.2

If the Required Action and the associated Completion Time is not met for any EFW Initiation Function, 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 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

E.1, E.2.1, and E.2.2

If the Required Actions and associated Completion Times are not met for the Main Steam Line Isolation Function, the unit must be placed in a MODE or condition in which the requirement does not apply. This is initiated by placing the unit in MODE 3 within 6 hours and, either reducing SG pressure to less than 750 psig, or closing and deactivating all associated valves, i.e., the valves which EFIC would close if it were to actuate while OPERABLE. 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 3.3.12.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the trains can perform their intended functions. However, for Main Steam Line Isolation and EFW Initiation, the test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of valves associated with Main Steam Line Isolation or EFW Initiation during testing at power. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 31 days is based on operating experience with regard to channel OPERABILITY that demonstrates the rarity of more than one train failing within the same 31 day interval.~~

REFERENCES

1. IEEE-279-1971, April 1972.

ACTIONS (continued)

A.1 (continued)

Condition A can be thought of as equivalent to failure of a single train of a two train safety system (e.g., the safety function can be accomplished, but a single failure cannot be taken). Thus, the Completion Time of 72 hours has been chosen to be consistent with Completion Times for restoring one inoperable ESF System train.

The EFIC System has not been analyzed for failure of both trains of the same Function. Consequently, any combination of failures in both trains A and B is not covered by Condition A and must be addressed by entry into LCO 3.0.3.

B.1 and B.2

If Required Action A.1 and its associated Completion Time is not met for the EFW Initiation Function, 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 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, and C.2.2

If the Required Actions and associated Completion Times are not met for the Main Steam Line Isolation Function, the unit must be placed in a MODE or condition in which the requirement does not apply. This is initiated by placing the unit in MODE 3 within 6 hours and, either reducing SG pressure to less than 750 psig, or closing and deactivating all associated valves, i.e., the valves which EFIC would close if it were to actuate while OPERABLE. 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 3.3.13.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the trains can perform their intended functions. This test verifies Main Steam Line Isolation and EFW Initiation automatic actuation logics are functional. This test simulates the required inputs to the logic circuit and verifies successful operation of the automatic actuation logic. The test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of valves associated with Main Steam Line Isolation or actuation of EFW during testing at power. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 31 days is based on operating experience with regard to channel OPERABILITY, which has demonstrated the rarity of more than one channel failing within the same 31 day interval.~~

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

SR 3.3.14.1 is the performance of a CHANNEL FUNCTIONAL TEST every 31 days. This test demonstrates that the EFIC vector logic performs its function as desired. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on operating experience with respect to channel OPERABILITY that demonstrates the rarity of more than one channel failing within the same 31-day interval.

REFERENCES

None.

ACTIONS (continued)

F.1 (continued)

The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels. The Special Report is to be submitted in accordance with 10 CFR 50.4 within 30 days of entering Condition F.

Both the RCS Hot Leg Level and the Reactor Vessel Level are methods of monitoring for inadequate core cooling.

The alternate means of monitoring the Reactor Building Area Radiation (High Range) consist of a combination of installed area radiation monitors and portable instrumentation.

The Completion Time of "Immediately" for Required Action F.1 identifies the start of the "clock" for submittal of the Special Report.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs apply to each PAM instrumentation Function in Table 3.3.15-1.

SR 3.3.15.1

Performance of the CHANNEL CHECK ~~once every 31 days~~ for each required instrumentation channel that is normally energized provides reasonable assurance for prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel with 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 drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared with similar unit instruments located throughout the unit. For the reactor building hi-range radiation monitor, the CHANNEL CHECK should also note the detector's response to the keep alive source.

Agreement criteria are determined by the unit 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 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 is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are, where practical, verified to be reading at the bottom of the range and not failed downscale.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.15.1 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency is based on unit operating experience that demonstrates channel failure is rare.~~ The CHANNEL CHECK supplements less formal but more frequent checks of channels during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.15.2

A CHANNEL CALIBRATION ~~is performed every 18 months.~~ CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. This test verifies the channel responds to measured parameters within the necessary range and accuracy.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult, and there is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices, with minimal drift. Finally, the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

For the Reactor Building Area Radiation instrumentation, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr, and a one point calibration check of the detector below 10 R/hr with a gamma source.

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detector (RTD) sensors is accomplished by an in-place cross calibration that compares the other sensing elements with the recently installed sensing element.

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors is accomplished by an in-place cross calibration that compares the other sensing elements with the recently installed sensing element.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of at least an 18-month calibration interval in the determination of the magnitude of equipment drift.~~

REFERENCES

1. SAR, Table 7-11A.
2. Regulatory Guide 1.97.
3. NUREG-0737, 1979.

ACTIONS (continued)

C.1 and C.2

If the CREVS cannot be placed into the emergency recirculation mode while in MODE 1, 2, 3, or 4, actions must be taken to minimize the chances of an accident that could lead to radiation releases. The unit must be placed in at least MODE 3 within 6 hours, with a subsequent cooldown to MODE 5 within 36 hours. This places the reactor in a low energy state that allows greater time for operator action if habitation of the control room is precluded. 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.

D.1 and D.2

Required Action D.1 is the same as discussed earlier for Condition A, except for Completion Time. If the CREVS cannot be placed into recirculation mode while moving irradiated fuel assemblies, then Required Action D.2 suspends actions that could lead to an accident that could release radioactivity resulting from a fuel handling accident.

Required Action D.2 places the irradiated fuel in a safe and stable configuration in which it is less likely to experience an accident that could result in a release of radioactivity. The irradiated fuel must be maintained in these conditions until the automatic isolation capability is returned to operation or when manual action places one train of the CREVS into the emergency recirculation mode. The Completion Time of "Immediately" is consistent with the urgency of the situation and accounts for the high radiation function, which provides the only automatic Control Room Isolation function capable of responding to radiation release due to a fuel handling accident. The Completion Time does not preclude placing any fuel assembly into a safe position before ceasing any such movement.

Note that in certain circumstances, such as fuel handling in the fuel handling area during power operation, both Condition A or B and Condition D may apply in the event of channel failure(s).

SURVEILLANCE REQUIREMENTS

SR 3.3.16.1

Performance of a CHANNEL CHECK for the Control Room Isolation - High Radiation actuation instrumentation ~~once every 12 hours~~ provides reasonable assurance for prompt identification of a gross failure of instrumentation. Performance of the CHANNEL CHECK helps ensure that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Acceptance criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties. 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. If the channels are within the criteria, it is an indication that the channels are OPERABLE. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The Frequency is based on operating experience that demonstrates channel failure is rare.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.16.2

A Note allows a channel to be inoperable for up to 3 hours for surveillance testing without entering the associated Conditions and Required Actions, although during this time period it cannot actuate a control room isolation. This is based on the average time required to perform channel surveillance. It is not acceptable to remove channels from service for more than 3 hours to perform required surveillance testing without declaring the channel inoperable.

SR 3.3.16.2 is the performance of a CHANNEL FUNCTIONAL TEST ~~once every 31 days~~ to ensure that the channels can perform their intended functions. This test verifies the capability of the instrumentation to provide the automatic Control Room Isolation. Any setpoint adjustment shall be consistent with the setpoint requirements.

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 31 day Frequency is based on operating experience which indicates that the instrumentation usually passes the CHANNEL FUNCTIONAL TEST when performed on a monthly basis.~~

SR 3.3.16.3

This SR requires the performance of a CHANNEL CALIBRATION to ensure that the instrument channel remains operational with the correct setpoint.

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The 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 drifts between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATION must be performed consistent with the setpoint requirements.

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on the assumption of at least an 18 month calibration interval in the determination of the magnitude of equipment drift and is consistent with the typical refueling cycle.~~

REFERENCES

1. ANO-2 SAR, Section 6.4.
 2. 10 CFR 50.36.
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APPLICABILITY (continued)

The Note indicates the limit on RCS pressure may be exceeded during short term operational pressure transients resulting from a THERMAL POWER change > 5% RTP per minute. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, for transients initiated from power levels less than the Allowable Thermal Power, increased DNBR margin exists to offset the temporary pressure variations.

ACTIONS

A.1

Loop pressure and hot leg coolant temperature are controllable and measurable parameters. With one or both of these parameters not within the LCO limits, action must be taken to restore the parameters. RCS flow rate is not a controllable parameter and is not expected to vary during steady state operation. However, if the flow rate is below the LCO limit, the parameter must be restored to within limits or power must be reduced as required in Required Action B.1, to eliminate the potential for violation of the minimum DNBR limit.

The 2-hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, determine the cause for the off normal condition, and restore the readings within limits. The Completion Time is based on plant operating experience.

B.1

If the Required Action and associated Completion Time 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 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis assumptions.

The 6-hour Completion Time is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

The RCS pressure value specified is dependent on the number of pumps in operation and has been adjusted to account for the pressure difference between the core exit and the measurement location. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation is within safety analysis assumptions.~~

A Note has been added to indicate the pressure limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.2

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The 12-hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.~~

A Note has been added to indicate the temperature limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

SR 3.4.1.3

The ~~12-hour~~ Surveillance Frequency for RCS total flow rate is performed using the available flow indications. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The 12-hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.~~

SR 3.4.1.4

Measurement of RCS total flow rate ~~once every 18 months~~ allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is greater than or equal to the minimum required RCS flow rate specified in the COLR.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered or RCS flow characteristics may have been modified, which may have caused change of flow.~~

The Surveillance is modified by a Note that indicates the SR does not need to be performed until stable thermal conditions are established at higher power levels (i.e., $\geq 90\%$ RTP). The Note provides for measurement of the flow rate at normal operating conditions at power in MODE 1. The Surveillance may be performed at low power or in MODE 2 or below. However, at low or zero power conditions, the indications are less accurate and significant penalties for uncertainties may be necessary. Performance of the calorimetric heat balance at a high power level and normal operating conditions provides for the most accurate flow verification.

REFERENCES

1. SAR, Chapter 14.
2. SAR, Section 3A.6.
3. BAW-10179P-A, Rev. 6, August 2005.
4. 10 CFR 50.36.

APPLICABILITY

The reactor has been designed and analyzed to be critical in MODES 1 and 2 only with $T_{avg} \geq 525^{\circ}\text{F}$. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1 and MODE 2.

ACTIONS

A.1

With T_{avg} below 525°F , 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 MODE 3 in 30 minutes. Rapid reactor shutdown can be readily and practically achieved in a 30-minute period. The Completion Time reflects the ability to perform this Action and maintain the plant within the analyzed range. If T_{avg} can be restored within the 30 minute time period, shutdown is not required.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS average temperature is required to be verified periodically at or above 525°F ~~every 12 hours. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The SR to verify RCS average temperature every 12 hours takes into account indications that are continuously available to the operator in the control room and is consistent with other routine surveillances which are typically performed once per shift.~~ In addition, Operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

RCS T_{avg} is normally calculated as the average of the unit T_{hot} (hot temperature average of loops A and B) and the unit T_{cold} (cold temperature average of loops A and B). During operation with 3 RCPs in operation, T_{avg} is calculated as the average of the loop T_{hot} and loop T_{cold} in the loop that has 2 RCPs running.

REFERENCES

1. SAR, Chapter 14.
 2. 10 CFR 50.36.
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ACTIONS (continued)

C.1 and C.2 (continued)

If the required evaluation for continued operation cannot be accomplished within 72 hours, or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Actions C.1 and C.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions. However, if the favorable evaluation is accomplished while reducing pressure and temperature conditions, a return to power operation may be initiated without completing these Required Actions.

Pressure and temperature are reduced by bringing the unit to 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 MODE from full power conditions in an orderly manner and without challenging unit systems.

D.1 and D.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified acceptable by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished promptly in a controlled manner.

In addition to restoring operation to within limits, an evaluation is required to verify that the RCPB integrity remains acceptable. The evaluation must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analysis, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 10), may also be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition D is modified by a Note requiring Required Action D.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 D.1, is insufficient because higher than analyzed stresses may have occurred and may have affected RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1, SR 3.4.3.2, SR 3.4.3.3, and SR 3.4.3.4

Verification that operation is within the limits of the appropriate figure is required ~~every 30 minutes~~ when RCS pressure and temperature conditions are undergoing planned changes. The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.3.1, SR 3.4.3.2, SR 3.4.3.3, and SR 3.4.3.4 (continued)

~~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.~~

Surveillance for heatup, cooldown, or inservice hydrostatic testing may be discontinued when the definition given in the relevant unit procedure for ending the activity is satisfied.

The acceptable P/T combinations are below and to the right of the limit curves. The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation (as identified in Note 1 on each applicable Figure).

SR 3.4.3.1 is modified by a Note that requires this SR to be performed only during system heatup operations with fuel in the reactor vessel. This SR refers to Figure 3.4.3-1 which provides applicable heatup limitations, including reactor coolant pump (RCP) operating restrictions and allowable heatup rates. Figure 3.4.3-1 Note 2 identifies that when the decay heat removal system is operating with no RCPs operating, the indicated DHR system return temperature to the reactor vessel is the appropriate temperature indicator.

SR 3.4.3.2 is modified by a Note that requires this SR to be performed only during system cooldown operations with fuel in the reactor vessel. This SR refers to Figure 3.4.3-2 which provides applicable cooldown limitations, including reactor coolant pump (RCP) operating restrictions and allowable cooldown rates. During system cooldown operations with fuel in the reactor vessel, the RCPs are eventually removed from service. Figure 3.4.3-2 Note 2 identifies that when the decay heat removal system is operating with no RCPs operating, the indicated decay heat removal system return temperature to the reactor vessel is the appropriate temperature indicator. Figure 3.4.3-2 Note 2 also indicates that a maximum step temperature change of 25 °F is allowable when removing all RCPs from operation with the decay heat removal system operating. The step temperature change is defined as the reactor coolant temperature (prior to stopping all RCPs) minus the decay heat removal system return temperature to the reactor vessel (after stopping all RCPs). The step change of 25 °F is applicable only during transition from RCP operation to DHR. This step change must be included when determining the cooldown rate.

SR 3.4.3.3 is modified by a Note that requires this SR to be performed only during system heatup and cooldown operations with no fuel in the reactor vessel. This SR refers to Figure 3.4.3-3. These curves are used during inservice hydrostatic testing that is performed in a defueled condition. The Notes on Figure 3.4.3-1 and Figure 3.4.3-2 are applicable to heatups and cooldowns performed within these limits.

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 helps ensure that forced flow is providing heat removal while maintaining the margin to DNB. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 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.

REFERENCES

1. SAR, Chapters 14 and 3A.
 2. BAW-10103A, Revision 3, July 1977.
 3. 10 CFR 50.36.
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ACTIONS

A.1

If one RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the RCS loop to OPERABLE status within a Completion Time of 72 hours. This time allowance is a justified period to be without the redundant non-operating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the DHR System. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to achieve cooldown and depressurization from the existing unit conditions and without challenging unit systems.

C.1 and C.2

If no RCS loop is OPERABLE or a required RCS loop is not in operation, (no RCS loop is required to be in operation provided the conditions of the Note in the LCO section are met), all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be immediately suspended. Action to restore one RCS loop to operation shall be immediately initiated and continued until one RCS loop is restored to operation and to OPERABLE status. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects 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 loop (and pump) is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 12-hour interval 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.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.2

Verification that each required RCP is OPERABLE ensures that an 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 each required pump that is not in operation. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. 10 CFR 50.36.
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ACTIONS (continued)

B.1 and B.2

If no RCS or DHR loops are OPERABLE or a required loop is not in operation (no loop is required to be in operation provided the conditions of the Note in the LCO section are met), all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or DHR 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 coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must continue until one loop is restored to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance requires verification ~~every 12 hours~~ of the required DHR or RCS loop in operation to ensure forced flow is providing decay heat removal. Verification includes flow rate, temperature, or pump status monitoring. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 12-hour interval 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.~~

SR 3.4.6.2

Verification that each required pump is OPERABLE ensures that an RCS or DHR 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 each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 7 days is considered reasonable in view of other administrative controls and has been shown to be acceptable by operating experience.~~

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. 10 CFR 50.36.
2. BAW-2441-A, Revision 2, Risk Informed Justification for LCO End-State Changes, September 2006.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification ~~every 12 hours~~ that the required DHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 12-hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation.~~ In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ 20 inches ensures that redundant heat removal paths are available if the second DHR loop is not OPERABLE. If both DHR loops are OPERABLE, this Surveillance is not needed. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 12-hour Frequency has been shown by operating practice to be sufficient to regularly assess RCS loop status.~~

SR 3.4.7.3

Verification that each required DHR pump is OPERABLE ensures that a DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. If the secondary side water level is ≥ 20 inches in both SGs, this Surveillance is not needed. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. 10 CFR 50.36.
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ACTIONS (continued)

B.1, B.2, and B.3

If no required loop is OPERABLE or the required loop is not in operation, except as provided by Note 1 in the LCO, the Required Action requires immediate suspension of all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 or reduction of RCS water inventory and requires initiation of action to immediately restore one DHR loop to OPERABLE status and operation. The Required Action for restoration does not apply to the condition of both loops not in operation when the exception Note in the LCO is in force. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operations for decay heat removal. The action to restore must continue until one loop is restored.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance requires verification ~~every 12 hours~~ that at least one loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status.~~

SR 3.4.8.2

Verification that each required pump is OPERABLE ensures that redundancy for heat removal is provided. The requirement also ensures that a DHR 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 each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. Generic Letter 88-17, October 17, 1988.
2. 10 CFR 50.36.

ACTIONS (continued)

B.1 and B.2

If the water level cannot be restored, reducing core power constrains heat input effects that drive pressurizer insurge that could result from an anticipated transient. By shutting down the reactor and reducing reactor coolant temperature to at least MODE 3, the potential thermal energy of the reactor coolant mass for mass and energy releases is reduced.

Six hours is a reasonable time based upon operating experience to reach MODE 3 from full power in an orderly manner and without challenging unit systems. Further pressure and temperature reduction to MODE 4 with RCS temperature ≤ 259 °F places the unit into a MODE where the LCO is not applicable. The 24 hour Completion Time to reach the non-applicable MODE is reasonable based upon operating experience.

C.1

If the required pressurizer heaters are inoperable, restoration is required in 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power will not occur in this period. Pressure control may be maintained during this time using non-ES bus powered heaters.

D.1 and D.2

If the Required Action and associated Completion Time are not met, 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 MODE 3 within 6 hours and to MODE 4 within the following 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable based on operating experience to achieve power reduction from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that pressurizer water level be maintained below the upper limit to provide a minimum space for a steam bubble. The value specified for pressurizer level does not contain an allowance for instrument error. Therefore, additional allowances for instrument uncertainties must be provided in the implementing procedures. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 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.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.9.2

The SR requires sufficient pressurizer heaters which are connected to an ES bus verified to be capable of providing the required capacity (this may be performed by testing the power supply output and by performing an electrical check on heater element continuity and resistance). ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 18 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.~~

REFERENCES

1. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.
 2. 10 CFR 50.36, Technical Specifications.
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ACTIONS (continued)

C.1 and D.1 (continued)

These Completion Times also consider these activities can be accomplished in these time periods. A limiting LTOP event is not likely in those times.

Some ERV testing or maintenance can only be performed at unit shutdown. Such activity is permitted if Required Action D.1 is taken to compensate for required ERV unavailability.

E.1

With the LTOP requirements not met for any reason other than cited in Condition A through D, action must be initiated to restore compliance immediately. The immediate Completion Time reflects the urgency of quickly proceeding with the Required Actions.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Verification of the pressurizer level at ≤ 180 inches by observing control room or other indications ensures that the unit is not in a water solid condition and that a cushion of sufficient size is available to reduce the rate of pressure increase from potential transients (Ref. 11). This parameter does not contain allowances for instrument error.

The 30-minute Surveillance Frequency during heatup and cooldown must be performed for the LCO Applicability period when temperature changes can cause pressurizer level variations. This Frequency may be discontinued when these evolutions are complete, as defined in unit procedures. ~~Thereafter, the Surveillance is required at 12-hour intervals.~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. These Frequencies are shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.~~

SR 3.4.11.2 and SR 3.4.11.3

Verifications must be performed that the HPI is deactivated, and each pressurized CFT is isolated. These Surveillances ensure the minimum coolant input capability will not create an RCS overpressure condition to challenge the LTOP. ~~The Surveillances are required at 12-hour intervals.~~

~~The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program. The 12-hour intervals are shown by operating practice to be sufficient to assess coolant input capability and verify operation within the safety analysis.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.4

OPERABLE pressure relief capability must be provided to prevent overpressurization due to inadvertent full makeup system operation. Such a vent keeps the pressure from full makeup flow within the LCO limit. OPERABLE pressure relief capability may be provided by an OPERABLE ERV, or by depressurizing the RCS and providing an alternate RCS vent path.

For the ERV to be considered OPERABLE, its block valve must be open, its lift setpoint must be set at ≤ 508 psig, testing must have proven its ability to open at that setpoint, and motive power must be available to the two valves and their control circuits. The parameter value of 508 psig does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

With the RCS depressurized, acceptable alternate vent paths include: a) removing a pressurizer safety valve; b) locking the ERV in the open position and disabling its block valve in the open position; c) removing a SG primary manway; c) removing a SG primary hand hole cover; d) removing all control rod drive top closure assemblies (excluding reactor vessel level probe); and e) removing a pressurizer manway.

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. For a vent path not locked open, the Frequency is every 12 hours. For a locked open vent path, the required Frequency is every 31 days.~~

~~The Frequency intervals are considered adequate based on operating practice to determine adequacy of pressure relief capability and verify operation within the safety analysis.~~

SR 3.4.11.5

~~The performance of a CHANNEL CALIBRATION is required every 18 months.~~ The CHANNEL CALIBRATION for the LTOP ERV opening logic, including the ERV setpoint, ensures that the ERV will be actuated at the appropriate RCS pressure by verifying the accuracy of the instrument string. The calibration can only be performed in shutdown.

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18 month Frequency considers a typical refueling cycle and industry accepted practice.~~

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

SR 3.4.12.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant ~~at least once every 7 days~~. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in the noble gas specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. ~~The 7 day Frequency considers the low probability of a gross fuel failure during this time.~~ The Surveillance is modified by a note requiring performance in MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq 500^{\circ}\text{F}$. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 7 day Frequency is based on the low probability of a gross fuel failure during that time period.~~

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.12.1 calculation. If a specific noble gas nuclide listed in the definition of DEX is not detected, it should be assumed to be present at the minimum detectable activity.

SR 3.4.12.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days.~~

REFERENCES

1. 10 CFR 50.67.
 2. ANO-1 Operating License Amendment 238 (1CNA100901), dated October 21, 2009.
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SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.1 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~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 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.16, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 8. 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 RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~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.~~ During normal operation the primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with EPRI Guidelines (Ref. 8).

REFERENCES

1. SAR, Section 1.4, GDC 30.
2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.
3. Information Submittal - Comparison of ANO-1 RCS Leak Detection Systems to Regulatory Guide 1.45 (1CAN108607), dated October 14, 1986.
4. SAR, Chapter 14.
5. SAR, Section 4.2.3.8.
6. 10 CFR 50.36.
7. NEI 97-06, "Steam Generator Program Guidelines."

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.2, SR 3.4.14.3, SR 3.4.14.4, and SR 3.4.14.5

Verifying that the DHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not over pressurize the DHR system. The interlock(s) that prevent the valves from being opened and that close the valves are designed to protect the DHR System from gross overpressurization. Although the specified values include certain process measurement uncertainties, additional allowances for instrument uncertainty are contained in the implementing procedures.

The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and on the potential for an unplanned transient if the Surveillance was performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.~~

REFERENCES

1. "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," issued April 20, 1981.
 2. NUREG-75/014, Reactor Safety Study, Appendix V, October 1975.
 3. NUREG-0677, The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes, May 1980.
 4. 10 CFR 50.36.
 5. ASME OM Code 2004 Edition through 2006 Addenda and Code Case OMN-20 (Inservice Test Frequency).
 6. 10 CFR 50.55a(f).
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ACTIONS (continued)

D.1 and D.2 (continued)

Required Action D.2 is modified by a second Note. Note 2 states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

E.1

With both required monitors inoperable, no indicated means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required reactor building atmosphere radioactivity monitor. The check gives reasonable confidence that each channel is operating properly. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.~~

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required reactor building atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm function and relative accuracy of the instrument string. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 92 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.~~

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the required RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside the reactor building. ~~The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Additionally, operating experience has shown this Frequency is acceptable.~~

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification ~~every 12 hours~~ that each CFT isolation valve is fully open ensures that the CFTs are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in accident analysis assumptions not being met. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program~~~~A 12 hour Frequency is considered reasonable in view of administrative controls that ensure that a mispositioned isolation valve is unlikely.~~

SR 3.5.1.2 and SR 3.5.1.3

Verification ~~every 12 hours~~ of each CFT's nitrogen cover pressure and the borated water volume is sufficient to ensure adequate injection during a LOCA. ~~The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program~~~~Due to the static nature of these parameters, a 12 hour Frequency usually allows the operator to identify changes before the limits are reached. Operating experience has shown that this Frequency is appropriate for early detection and correction of off normal trends.~~

SR 3.5.1.4

~~This Surveillance~~ ~~once every 31 days is reasonable to~~ verify that the CFT boron concentration is within the required limits, because the static nature of this parameter limits the ways in which the concentration can be changed. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program~~~~The Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.~~

Sampling of the affected CFT within 12 hours after a 0.2 ft volume increase will identify whether inleakage from the RCS has caused a reduction in boron concentration to below the required limit. The 0.2 ft increase represents approximately 102 gallons increase in volume. It is not necessary to verify boron concentration if the added water inventory is from a borated water source of known concentration ≥ 2270 ppm, such as the borated water storage tank (BWST), because the water is within CFT boron concentration requirements. Similarly, it would not be necessary to sample the CFT following inventory additions from the CFT makeup tank if sampling has determined that the added inventory had a boron concentration within the CFT requirements. This is consistent with the recommendations of NUREG-1366 (Ref. 4).

SR 3.5.1.5

Removing power from each CFT isolation valve operator ensures that an active failure could not result in the undetected closure of a CFT motor operated isolation valve coincident with a LOCA. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program~~~~Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that the power is removed.~~

SURVEILLANCE REQUIREMENTS

For inservice testing periods up to and including 2 years, Code Case OMN-20 provides an allowance to extend the inservice testing periods by up to 25%. For inservice testing periods of greater than 2 years, OMN-20 allows an extension of up to 6 months.

SR 3.5.2.1

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 nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 31-day Frequency is appropriate because the valves are operated under administrative control, and an inoperable valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.~~

SR 3.5.2.2

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME OM Code (Ref. 7). This testing confirms component OPERABILITY, trends performance, and detects incipient failures by indicating abnormal performance. SRs are specified in the INSERVICE TESTING PROGRAM, which encompasses the ASME OM Code.

SR 3.5.2.3

This SR demonstrates that each automatic ECCS valve actuates to the required position on an actual or simulated ESAS signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18-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 ESAS testing, and equipment performance is monitored as part of the INSERVICE TESTING PROGRAM.

SR 3.5.2.4

The intent of this SR is to verify that the ECCS pumps are capable of automatically starting on an ESAS signal. Because of the system design configuration and the limitations imposed on pump operation during the unit conditions when this test would be conducted, this verification must be conducted through a series of sequential, overlapping or total steps in order to demonstrate functionality.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.4 (continued)

SR 3.5.2.4 demonstrates that each ECCS pump would be capable of starting by verifying that its breaker closes on receipt of an actual or simulated ESAS signal. SR 3.5.2.4 works in conjunction with the INSERVICE TESTING PROGRAM (SR 3.5.2.2) which periodically verifies the ability of the pumps to start and operate within limits, and the ESAS actuation logic testing which periodically verifies the ability of the ESAS to sense, process and generate an actuation signal.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.~~

SR 3.5.2.5

Periodic inspections of the reactor building sump suction inlet ensure that it is unrestricted and stays in proper operating condition. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18 month Frequency is based on the need to perform this Surveillance during a unit outage. Operating experience has shown this Frequency to be acceptable to detect abnormal degradation.~~

REFERENCES

1. SAR, Section 6.
2. Letter from A. C. Thadani (NRC) to P. S. Walsh (BWO) dated March 9, 1993.
3. 10 CFR 50.46.
4. SAR, Section 14.2.2.5.2.
5. 10 CFR 50.36.
6. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
7. ASME OM Code 2004 Edition through 2006 Addenda and Code Case OMN-20 (Inservice Test Frequency).
8. Condition Report CR-ANO-1-2009-0997.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

Verification ~~every 24 hours~~ that the BWST water temperature is within the specified temperature band ensures that the fluid will not freeze and that the fluid temperature will not be hotter than assumed in the safety analysis. These parameter values are considered to be nominal values and do not contain an allowance for instrument uncertainty. No additional allowances for instrument uncertainty are required to be included in the implementing procedures. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 24-hour Frequency is sufficient to identify a temperature change that would approach either temperature limit.~~

The SR is modified by a Note that requires the Surveillance to be performed only when ambient air temperatures are outside the operating temperature limits of the BWST. With ambient temperatures within this band, the BWST temperature should not exceed the limits.

SR 3.5.4.2

Verification ~~every 7 days~~ that the BWST level is ≥ 38.4 feet and ≤ 42 feet ensures that a sufficient initial supply is available for injection and to support continued ECCS pump operation on recirculation. These levels correspond to volumes of approximately 375,096 gallons and 405,090 gallons, respectively. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Since the BWST level is normally stable, a 7-day Frequency has been shown to be appropriate through operating experience.~~

SR 3.5.4.3

Verification ~~every 7 days~~ that the boron concentration of the BWST fluid is ≥ 2270 ppm and ≤ 2670 ppm ensures that the reactor will remain adequately shutdown following a LOCA. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Since the BWST level is normally stable, a 7-day sampling Frequency is appropriate and has been shown to be acceptable through operating experience.~~

REFERENCES

1. SAR, Section 6.1.
2. Letter from A. C. Thadani (NRC) to P. S. Walsh (BWO) dated March 9, 1993.
3. 10 CFR 50.36.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident reactor building pressure, closure of either door will support the reactor building OPERABILITY. Thus, the door interlock feature supports the reactor building OPERABILITY while the air lock is being used for personnel transit in and out of the reactor building. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur.

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the reactor building air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 18 months. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage, and the potential for loss of reactor building OPERABILITY if the Surveillance were performed with the reactor at power. The 18 month Frequency for the interlock is justified based on generic operating experience. The 18 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not expected to be challenged during use of the airlock.~~

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
 2. SAR, Chapter 14.
 3. SAR, Chapter 5.
 4. 10 CFR 50.36.
 5. BAW-2441-A, Revision 2, Risk Informed Justification for LCO End-State Changes, September 2006.
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ACTIONS (continued)

D.1 and D.2

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 7). The release of stored energy to the Reactor Building in the event of an accident in MODE 4 is substantially less than the energy release assumed due to an accident at power. Therefore, the challenge to containment isolation valves is substantially reduced. Because of the reduction in RCS pressure and temperature in MODE 4, the likelihood of an event is also reduced. In addition, there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms in MODE 4 than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action D.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 24-inch reactor building purge isolation valve in the purge system supply and exhaust is required to be periodically verified closed ~~at 31 day intervals~~. This Surveillance is designed to ensure that a gross breach of the reactor building is not caused by an inadvertent opening of a reactor building purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the closed position during MODES 1, 2, 3, and 4. A reactor building purge valve that is closed must have motive power to the valve operator removed. This can be accomplished by removing the valve handswitch key. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program~~The Frequency is consistent with other reactor building isolation valves discussed in SR 3.6.3.2.~~

SR 3.6.3.2

This SR requires verification that each reactor building isolation manual valve and blind flange located outside the reactor building 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 reactor building 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 reactor building isolation valves outside the reactor building and capable of being mispositioned are in the correct position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program~~Since verification of valve position for the reactor building isolation valves outside the reactor building is relatively easy, the 31 day Frequency was chosen to provide added assurance of the correct positions.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.5

Automatic reactor building isolation valves close on a reactor building isolation signal to prevent leakage of radioactive material from the reactor building following a DBA. This SR ensures that each automatic reactor building isolation valve will actuate to its isolation position on a reactor building isolation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

REFERENCES

1. SAR, Chapter 5.
 2. SAR, Chapter 14.
 3. 10 CFR 50.36.
 4. SAR, Table 5-1.
 5. Generic Letter 91-08, Removal of Component Lists from Technical Specifications.
 6. Condition Report CR-ANO-1-2010-2515.
 7. BAW-2441-A, Revision 2, Risk Informed Justification for LCO End-State Changes, September 2006.
 8. ASME OM Code 2004 Edition through 2006 Addenda and Code Case OMN-20 (Inservice Test Frequency).
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ACTIONS (continued)

B.1 and B.2 (continued)

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 5). The release of stored energy to the Reactor Building in the event of an accident in MODE 4 is substantially less than the energy release assumed due to an accident at power. Therefore, the challenge to the containment systems due to an increase in containment pressure is substantially reduced. Because of the reduction in RCS pressure and temperature in MODE 4, the likelihood of an event is also reduced. In addition, there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms in MODE 4 than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that the reactor building pressure is within limits ensures that operation remains within the limits assumed in the ECCS and the reactor building analyses. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 12-hour Frequency of this SR was developed after taking into consideration operating experience related to trending of the reactor building pressure variations during the applicable MODES. Furthermore, the 12-hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal reactor building pressure condition.~~

REFERENCES

1. SAR, Chapter 14.
2. SAR, Chapter 5.
3. 10 CFR 50, Appendix K.
4. 10 CFR 50.36.
5. BAW-2441-A, Revision 2, Risk Informed Justification for LCO End-State Changes, September 2006.

ACTIONS (continued)

G.1

With two reactor building spray trains inoperable in MODE 1 or 2, or any combination of three or more reactor building spray and reactor building cooling trains inoperable in MODE 1 or 2, or one required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4, then LCO 3.0.3 must be entered immediately. The first part of this Condition addresses the loss of Spray Additive System support which would result from two inoperable reactor building spray trains in MODE 1 or 2. The second part of this Condition considers the loss of adequate reactor building cooling capacity in MODE 1 or 2 which would result from the loss of three or more of the four RB spray and RB cooling trains. Finally, the third part of this Condition addresses loss of reactor building cooling capability in MODES 3 and 4 when only one train of RB spray and one train of RB cooling are required.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the reactor building spray flow path provides assurance that the proper flow paths will exist for the Reactor Building 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 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, through a system walkdown or control room indication, that those valves outside the reactor building and capable of potentially being mispositioned are in the correct position. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)

SR 3.6.5.2

Operating each required reactor building cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. This SR is performed by starting (unless operating) each operational cooling fan from the control room. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The 31 day Frequency was developed considering the known reliability of the fan units and controls, the redundancy available, and the low probability of a significant degradation of the reactor building cooling trains occurring between surveillances and has been shown to be acceptable through operating experience.~~

SR 3.6.5.3

Verifying that a service water flow rate of 1200 gpm is provided to each required reactor building cooling train provides assurance that the original design flow rate is being achieved and that the service water flow rate is not degrading (Ref. 3). Assurance that the flow doesn't degrade by biological fouling between surveillances is provided by the addition of a biocide to the Service Water System whenever the service water temperature is between 60 °F and 80 °F. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The Frequency was developed considering the known reliability of the system, the redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.4

Verifying that each required reactor building 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. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow rate are within $\pm 10\%$ of a point on the pump head curve. Flow and differential pressure are measured during normal tests of centrifugal pump performance required by the ASME OM Code (Ref. 5). Since the Reactor Building Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms the discharge pressure and flow rate are within $\pm 10\%$ of a point on the pump head curve and is indicative of overall pump performance. Such inservice tests confirm component OPERABILITY, trend performance, and may detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

For inservice testing periods up to and including 2 years, Code Case OMN-20 provides an allowance to extend the inservice testing periods by up to 25%. For inservice testing periods of greater than 2 years, OMN-20 allows an extension of up to 6 months.

SR 3.6.5.5 and SR 3.6.5.6

These SRs require verification that each automatic reactor building spray valve actuates to its correct position and that each reactor building spray pump starts upon receipt of an actual or simulated actuation signal. The SRs are considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. During testing of the spray pump, the reactor building isolation valve in the spray line is closed with its breaker open to prevent spraying the reactor building. After spray pump performance is verified, the pump is stopped. Its breaker is racked down to prevent restart. Power is then restored to the reactor building isolation valve for valve testing. ~~The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program. The 18-month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

SR 3.6.5.7

This SR requires verification by control board indication that each required reactor building cooling train actuates upon receipt of an actual or simulated actuation signal. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18-month Frequency has been shown to be acceptable through operating experience. See SR 3.6.5.5 and SR 3.6.5.6, above, for further discussion of the basis for the 18-month Frequency.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.1 (continued)

This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or control room indication, that those valves outside the reactor building capable of potentially being mispositioned are in the correct position. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)

SR 3.6.6.2

To provide the most effective iodine removal, the reactor building spray should be an alkaline solution. Since the BWST contents are normally acidic, the NaOH tank must provide a sufficient volume of NaOH to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The NaOH tank solution minimum volume of 9000 gallons corresponds to a tank level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 6.0 wt%. This parameter does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures. The minimum NaOH tank volume preserves the required NaOH solution contribution from the tank to the post-LOCA minimum sump level. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The 184 day Frequency is based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations).~~ Tank level is also indicated and alarmed in the control room, such that there is a high confidence that a substantial change in level would be detected.

SR 3.6.6.3

This SR provides verification of the NaOH concentration in the NaOH tank and is sufficient to ensure that the spray solution being injected into the reactor building is at the correct pH level. The concentration of NaOH in the NaOH tank must be determined by chemical analysis. This parameter is considered to be a nominal value; therefore, additional allowance for instrument uncertainty need not be applied. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The 184 day Frequency is sufficient to ensure that the concentration of NaOH in the tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.~~

SR 3.6.6.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.1 (continued)

This test is normally conducted in MODE 3, with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was generated.

For inservice testing periods up to and including 2 years, Code Case OMN-20 provides an allowance to extend the inservice testing periods by up to 25%. For inservice testing periods of greater than 2 years, OMN-20 allows an extension of up to 6 months.

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of MSIV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.~~

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.2.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when SG pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low steam generator pressure during a unit shutdown.

REFERENCES

1. SAR, Section 14.2.
 2. SAR, Section 7.1.4.
 3. 10 CFR 50.36.
 4. ASME OM Code 2004 Edition through 2006 Addenda and Code Case OMN-20 (Inservice Test Frequency).
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SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve is as specified in the INSERVICE TESTING PROGRAM.

The MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve isolation time is assumed in the accident and reactor building analyses. This Surveillance is normally performed prior to returning the unit to power operation, e.g., during MODE 3, following a refueling outage. The MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves are not tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The Frequency for this SR is in accordance with the INSERVICE TESTING PROGRAM.

For inservice testing periods up to and including 2 years, Code Case OMN-20 provides an allowance to extend the inservice testing periods by up to 25%. For inservice testing periods of greater than 2 years, OMN-20 allows an extension of up to 6 months.

SR 3.7.3.2

This SR verifies that each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)

[The Frequency for this SR is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.](#)

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.3.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when the SG pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low SG pressure during a unit shutdown.

LCO (continued)

Monitoring the specific activity of the secondary coolant ensures that, when secondary specific activity limits are exceeded, appropriate actions are taken, in a timely manner, to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are at low pressure and primary to secondary LEAKAGE is minimal. Therefore, secondary specific activity is not a concern.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits 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 3.7.4.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis assumptions. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the analysis assumptions are met. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 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.~~

REFERENCES

1. 10 CFR 50.67.
2. Safety Evaluation Report for ANO-1 License Amendment No. 2, 1CNA057502, dated May 9, 1975.

ACTIONS (continued)

E.1

In MODE 4, either the steam generator loops or the DHR loops can be used to provide heat removal, which is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With the required EFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation. Correct alignment for automatic valves may be other than the post-accident position provided the valve is otherwise OPERABLE. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since those 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 Surveillance 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.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 31 day Frequency is based on the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded below the established acceptance criteria during the cycle. Flow and differential head are indicators of pump performance required by the ASME OM Code (Ref. 5). Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this test may be performed on a test flow path.

This test 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 as discussed in the ASME OM Code (Ref. 5) and the INSERVICE TESTING PROGRAM satisfies this requirement.

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

For inservice testing periods up to and including 2 years, Code Case OMN-20 provides an allowance to extend the inservice testing periods by up to 25%. For inservice testing periods of greater than 2 years, OMN-20 allows an extension of up to 6 months.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR verifies that EFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an EFIC system signal by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18-month Frequency is also acceptable based on operating experience and design reliability of the equipment.~~

This SR is modified by a Note which states that the SR is not required to be met in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.

SR 3.7.5.4

This SR verifies that each EFW pump starts in the event of any accident or transient that generates an EFIC signal. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~

This SR is modified by a Note which states that the SR is not required to be met in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.

SR 3.7.5.5

This SR ensures that the EFW system is properly aligned by verifying the position of manual valves in the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable in view of other administrative controls, such as SR 3.7.5.1, to ensure that the flow paths are OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified, following extended outages to determine no misalignment of manual valves has occurred. This SR ensures that the flow path from the QCST to the steam generator is properly aligned.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.6

This SR ensures that the EFW flow path to each steam generator is open and that water reaches the steam generators from the EFW system. This test is performed during shutdown to minimize thermal cycles to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater. The motor-driven EFW pump is specified because of its availability at the low steam generator pressure conditions that exist in the shutdown condition. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~

REFERENCES

1. SAR, Section 7.1.4.
 2. SAR, Section 10.4.8.
 3. NRC Letter dated January 12, 1981, (1CNA018103).
 4. 10 CFR 50.36.
 5. ASME OM Code 2004 Edition through 2006 Addenda and Code Case OMN-20 (Inservice Test Frequency).
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APPLICABILITY

In MODES 1, 2, 3, and 4 when a steam generator is being relied upon for heat removal, the QCST is required to be OPERABLE.

In MODE 4 when a steam generator is not being relied upon for heat removal, and in MODES 5 and 6, the QCST is not required because the EFW System is not required.

ACTIONS

A.1 and A.2

As an alternative to unit shutdown, the OPERABILITY of the backup water supply (SWS) should be verified within 4 hours and once every 12 hours thereafter. The OPERABILITY of the SWS backup feedwater supply must include verification, by administrative means, of the OPERABILITY of the SWS flow path to the EFW pumps. The QCST must be restored to OPERABLE status within 7 days because the backup supply may be performing this function in addition to its normal functions. The 4-hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7-day Completion Time is reasonable, based on an OPERABLE SWS backup water supply being available, and the low probability of an event occurring during this time period, requiring the use of water from the QCST.

B.1 and B.2

If the Required Action and associated Completion Times are not met, the unit must be placed in a MODE in which the LCO does not apply, with the DHR System in operation. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generators for heat removal, within 24 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

This SR verifies that the QCST contains the required volume of cooling water. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 12-hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the QCST 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 deviations in QCST levels.~~

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS 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 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 also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The 31-day Frequency is based on the existence of procedural controls governing valve operation, and ensures correct valve positions.~~

This SR is modified by a Note indicating that the isolation of components or systems supported by the SWS does not affect the OPERABILITY of the SWS. However, such isolation may render those components inoperable.

SR 3.7.7.2

The SR verifies proper automatic operation of the SWS valves. The SWS is a normally operating system that cannot be fully actuated as part of the normal testing. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18-month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

SR 3.7.7.3

This SR requires verification that the normally operating SWS pumps (A and C) automatically restart following restoration of power to the respective bus. In addition, the B SWS pump, normally in the standby condition, must be verified to start to support each SWS train for which it is expected to be aligned upon associated ES actuation (with time delay) with simulated failure of the normally operating pump for that train.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The 18-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at an 18-month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

ACTIONS (continued)

B.1 and B.2

If the ECP is inoperable, 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 3.7.8.1

This SR (together with SR 3.7.8.3 and SR 3.7.8.4) verifies that adequate long term (30 days) cooling inventory is available. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 24-hour Frequency is based on operating experience related to the trending of the ECP level during the applicable MODES.~~ This SR verifies that the ECP indicated water level is ≥ 5.2 ft, which is sufficient to ensure a water volume of 70 acre-feet when crediting operator action to initiate makeup to the ECP upon a loss of Dardanelle Reservoir event (described in the Applicable Safety Analyses section above). The 5.2-foot minimum level requirement includes measurement, calculation, and other uncertainties.

SR 3.7.8.2

This SR provides assurance that the heat sink for the SWS can dissipate the maximum accident or normal heat loads for 30 days following the design basis event. The temperature, measured at the point of discharge from the ECP, is considered a conservative average of total ECP conditions since solar gain, wind speed, and thermal current effects throughout the ECP will essentially be at equilibrium conditions under initial stagnant conditions. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 24-hour Frequency is based on operating experience related to the trending of the ECP temperature during the applicable MODES.~~ This SR verifies that the ECP average water temperature at the point of discharge from the ECP (i.e., SWS suction) is ≤ 100 °F.

This SR is modified by a Note indicating that the temperature monitoring is required to be performed only during the summer months (i.e., June 1 to September 30). During other periods of the year, the ECP temperature will not have the potential to reach the temperature limit.

SR 3.7.8.3

This SR (together with SR 3.7.8.1 and 3.7.8.4) verifies that adequate inventory exists to support long term (30 days) cooling. Soundings are performed to ensure the water volume is within limits and that the indicated water level is indicative of an equivalent water volume for accident mitigation. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 12-month Frequency reflects the gradual pace of degradation of the physical properties of the ECP.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.4

This SR (together with SR 3.7.8.1 and 3.7.8.3) verifies that adequate inventory exists to support long term (30 days) cooling. Visual inspections of the loose stone (riprap) placed on the banks of the ECP and of the concrete spillway are performed to ensure erosion, undercut caused by wave action, or any physical degradation is within acceptable limits to enable the ECP to fulfill its safety function. An engineering evaluation of any apparent changes in visual appearance or other abnormal degradation is performed within 7 days to determine OPERABILITY. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 12-month Frequency reflects the gradual pace of degradation of the physical properties of the ECP.~~

REFERENCES

1. SAR, Section 9.3.
 2. Regulatory Guide 1.27, Rev. 1, "Ultimate Heat Sink for Nuclear Power Plants," March 1974.
 3. 10 CFR 50.36.
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ACTIONS (continued)

F.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary (i.e., Condition B), the CREVS may not be capable of performing the intended function and a loss of safety function has occurred. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

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 on a monthly basis adequately checks this system by initiating flow through the HEPA filters and charcoal adsorbers. The CREVS is designed without heaters and need only be operated for ≥ 15 minutes to demonstrate the function of the system. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The 31-day Frequency is based on the known reliability of the equipment and two train redundancy available.~~

SR 3.7.9.2

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

SR 3.7.9.3

This SR verifies that the CREVS automatically isolates the CRE within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks on an actual or simulated actuation signal. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The Frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle.~~

SR 3.7.9.4

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.

ACTIONS (continued)

B.1 and B.2 (continued)

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

C.1 and C.2

During movement of irradiated fuel, if the Required Action and associated Completion Time of Condition A are not met, the OPERABLE CREACS train must be placed in operation immediately. This action ensures that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that could release radioactivity that might require the isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

During movement of irradiated fuel assemblies, with two CREACS trains inoperable, action must be taken to immediately suspend activities that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CREACS trains are inoperable in MODE 1, 2, 3, or 4, a loss of safety function has occurred, and LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 and SR 3.7.10.2

These SRs, in conjunction with periodic preventative maintenance activities, provide verification that the CREACS will maintain the control room temperature within acceptable bounds. In accordance with SR 3.7.10.1, each train is verified to start and operate for at least 1 hour while maintaining Control Room temperature within the specified limit. ~~The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program. The Frequencies (31 days and 18 months) are appropriate as periodic preventative maintenance activities are routinely performed and significant degradation of the CREACS is not expected over these time periods.~~

ACTIONS (continued)

B.1 (continued)

Preplanned measures should be available to address these concerns for intentional and unintentional entry into the Condition. The 24-hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24-hour Completion Time is a typically reasonable time to diagnose, plan and possible repair, and test most problems with the PRVS negative pressure boundary.

C.1 and C.2

If the Required Action and the associated Completion Time are not met, or with both PRVS trains inoperable, 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 3.7.11.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 31-day Frequency is based on known reliability of equipment and the two train redundancy available.~~

SR 3.7.11.2

This SR verifies that the required PRVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.11.3

This SR verifies that each PRVS train starts and operates on an actual or simulated actuation signal. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18-month Frequency is consistent with the guidance provided in Regulatory Guide 1.52 (Ref. 4).~~

REFERENCES

1. SAR, Section 6.5.
 2. SAR, Sections 14.2.2.5 and 14.2.2.6.
 3. 10 CFR 50.36.
 4. ~~Regulatory Guide 1.52, "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," Rev. 2, March 1978.~~
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APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool 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, immediate action must be taken to preclude the occurrence of an accident. With the spent fuel pool at less than the required level, the movement of fuel assemblies in the spent fuel pool is immediately suspended. This effectively precludes the occurrence of a fuel handling accident. In such a case, unit procedures control the movement of loads over the spent fuel. This does not preclude movement of a fuel assembly to a safe position.

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 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 that sufficient spent fuel pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 7 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.~~

During refueling operations, the level in the spent fuel pool is at equilibrium with that in the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.

REFERENCES

1. FSAR, Section 9.6.1.3.
2. FSAR, Section 9.4.
3. FSAR, Section 14.2.2.3.
4. Regulatory Guide 1.183.
5. 10 CFR 50.67.
6. 10 CFR 50.36

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that the concentration of boron in the spent fuel pool and cask loading pit is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.~~

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
 2. SAR, Section 14.2.2.3.
 3. Safety Evaluation Report, Section 2.1.3, License Amendment No. 76, April 15, 1983.
 4. 10 CFR 50.36.
 5. 10 CFR 50.68.
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SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with SAR, Section 1.4, GDC 18 (Ref. 7). Periodic component tests are supplemented by extensive functional tests during outages (under simulated accident conditions).

Where the SRs discussed herein specify “ready-to-load” a minimum output voltage of 4000 volts is applicable. This value allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The required minimum frequency for loading of the DG is 58.8 Hz (derived from Safety Guide 9); however, this value is not routinely monitored to be within limit within 15 seconds. Meeting minimum frequency is expected prior to the DG voltage reaching the required minimum. This is administratively confirmed on an 18-month interval.

SR 3.8.1.1

This SR ensures correct breaker alignment for each required offsite circuit to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The SR also verifies the indicated availability of three-phase AC electrical power from each required offsite circuit to the onsite distribution network. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 7-day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.~~

SR 3.8.1.2

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, this SR is modified by a Note to indicate that DG starts for this Surveillance may be preceded by an engine prelube period and followed by a warmup period prior to loading.

For the purposes of SR 3.8.1.2 testing with application of the Note, the DGs are started from standby conditions. Standby conditions for a DG means that the diesel engine oil is being continuously circulated and temperature is being maintained consistent with manufacturer recommendations. The signal initiating the start of the DG is varied from one test to another (start with handswitch at control room panel and at DG local control panel) to verify all starting circuits are OPERABLE.

SR 3.8.1.2 requires that the DG starts from standby conditions and achieves “ready-to-load” conditions (i.e., minimum voltage) within 15 seconds. The 15 second start requirement supports the assumptions of the design basis LOCA analysis in the SAR, Chapter 14 (Ref. 4).

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 31-day Frequency provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting full rated load. The load test is conducted at 90 to 100 percent of the continuous rating, which is considered to be 90 to 100 percent of the intended service rating, or ≥ 2475 kW and ≤ 2750 kW. These parameter values contain all necessary allowances for instrument uncertainty. No additional allowances for instrument uncertainty are required to be incorporated in the implementing procedures. A minimum run time of 60 minutes ensures stabilized engine temperatures, while minimizing the time that the DG is connected to the offsite source.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The 31-day Frequency for this Surveillance provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.~~

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance 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 (e.g., because of changing bus loads) do not invalidate this test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result 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 3.8.1.4

This SR provides verification that the level of fuel oil in the engine mounted day tank is being properly maintained. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel, when combined with the volume contained in one fuel oil storage tank, for not less than 3.5 days operation of one DG loaded to full capacity (Ref. 2).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~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 3.8.1.5

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 ~~and engine mounted~~ 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 Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 10).~~ The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 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.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, and the fuel delivery piping is not obstructed.

The design of the fuel transfer systems is such that pumps operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day tanks during DG monthly testing. ~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Therefore, a 31-day Frequency is specified to correspond to the interval for DG testing.~~

SR 3.8.1.7

Transfer of each 4.16 kV ES 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. Reference 1 requires that only one of the two offsite power circuits be capable of automatic transfer. The second (alternate) circuit must be capable of manual transfer, as a minimum. Typically, Startup Transformer No. 1 is aligned for automatic transfer and Startup Transformer No. 2 is aligned to allow manual transfer. In this alignment, the Surveillance verifies the automatic transfer of loads to Startup Transformer No. 1 and the manual transfer of loads to Startup Transformer No. 2. In the event that Startup Transformer No. 2 is aligned for automatic transfer and Startup Transformer No. 1 is aligned for manual transfer, the Surveillance verifies the automatic transfer of loads to Startup Transformer No. 2 and the manual transfer of loads to Startup Transformer No. 1.

For Startup Transformer No. 2, this test also demonstrates the selective load shedding interlock function. (Note: This load shedding function is only required when Startup Transformer No. 2 is selected for automatic transfer.) These features provide protection of required equipment from a sustained degraded grid voltage situation.

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18-month Frequency of the Surveillance takes into consideration the unit conditions required to perform the Surveillance (i.e., during refueling shutdown), and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

This SR is modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.7 (continued)

This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODES 1 or 2. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.8

This Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the non-essential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve “ready-to-load” conditions (i.e., minimum required voltage) within the specified time.

The DG auto-start time of 15 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads, e.g., the running service water pump(s), is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads can not actually be connected or loaded without undue hardship or potential for undesired operation. In lieu of actual demonstration of connection and loading of loads during this test, 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.

If the component start time delays are outside of those assumed by the SAR, component OPERABILITY and DG OPERABILITY must be evaluated.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~

This SR is modified by a Note. The reason for the Note is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

SR 3.8.1.9

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ES systems so that the fuel, RCS, and reactor building design limits are not exceeded.

SR 3.8.1.9 (continued)

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ES systems so that the fuel, RCS, and reactor building design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.7, during a loss of offsite power actuation test signal in conjunction with an ES actuation signal. This test is typically conducted by simulating an ESAS signal and either simultaneously or subsequently simulating a LOOP. In certain circumstances, many loads can not actually be connected or loaded without undue hardship or potential for undesired operation. For instance, DHR systems performing a DHR function are not desired to be interrupted from this mode of operation. In lieu of actual demonstration of connection and loading of loads during this test, 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.

For ANO-1, the High Pressure Injection (HPI) pump load is excluded from this DG test; although its auto start logic and load sequencing time are included in the test. HPI pump start under accident flow conditions can challenge LTOP protection considerations when the RCS is closed and challenge pump integrity due to excessive flow or inadequate suction when the RCS is open. The remaining loads that are auto-connected during this test in conjunction with design calculations are sufficient to demonstrate that the DG will perform as designed during a LOCA with coincident LOOP event.

Should the time intervals between two or more loads be reduced such that the interval is less than that assumed in the SAR, the associated DG is conservatively considered to be inoperable unless an evaluation of the condition shows the loading of the DG, with the reduced time interval, to be acceptable. If one or more time delays is inoperable (i.e., the associated component fails to load) or the time interval between two or more loads is greater than assumed in the SAR, then the associated component is considered inoperable, and the appropriate Condition for that component is entered.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.~~

This SR is modified by a Note. The reason for the Note is to minimize wear and tear on the DGs during testing. For the purpose of this testing with application of the Note, the DGs are started from standby conditions, that is, with the engine oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGs.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.4 is not required to be met since crediting manual start of the required DG provides sufficiently opportunity to ensure that the fuel oil transfer system is operating properly. SR 3.8.1.7 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.8 and SR 3.8.1.9 are not required to be met because they provide for testing of engineered safeguards actuation signals which are not required to be OPERABLE except in MODES 1, 2, 3, and 4. Automatic actuation and loading of the DGs is not assumed in MODES 5 and 6.

This SR is modified by two Notes. The reason for Note 1 is to preclude requiring the OPERABLE DG from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 4160 V ES bus or disconnecting a required offsite circuit during performance of this SR. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that this SR must be capable of being met, but actual performance is not required during periods when the DG and offsite circuit are required to be OPERABLE. When Note 1 is considered, SR 3.8.2.1 requires the following:

- SR 3.8.1.1 must be performed and met,
- SR 3.8.1.2 must be performed and met,
- SR 3.8.1.3 must be met, but does not have to be performed,
- SR 3.8.1.4 does not have to be performed or met,
- SR 3.8.1.5 must be performed and met,
- SR 3.8.1.6 must be performed and met,
- SR 3.8.1.7 does not have to be performed or met,
- SR 3.8.1.8 does not have to be performed or met, and
- SR 3.8.1.9 does not have to be performed or met.

Note 2 exempts the 15 second start acceptance criteria for SR 3.8.1.2. This allows the AAC DG power source, which does not have auto-start capability or start-time criteria, to be used in lieu of an emergency DG. In MODES 5 and 6, there is sufficient time to manually start a DG in the event the offsite power source is lost. The required DG must be capable of being started from standby conditions and achieving ready-to-load conditions. Although the time to reach ready-to-load conditions is not a part of the acceptance criteria, it is intended that for emergency DG tests, this time be trended to help determine if a condition exists that is degrading the starting capabilities of the DG.

[The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.](#)
Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.1 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~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 of fuel oil prior to addition to the storage tanks are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine operation. 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 sampling (and associated results) of new fuel and addition of new fuel oil to the storage tank(s) to exceed 31 days. The tests, limits, and applicable ASTM Standards for the tests listed in Specification 5.5.13, "Diesel Fuel Oil Testing Program," are as follows:

- a. Sample the new fuel oil in accordance with ASTM D4057-88 (Ref. 4); and
- b. Verify in accordance with the tests specified in ASTM D975-81 (Ref. 4) that the sample has:
 1. an absolute specific gravity at 60/60 °F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60 °F of $\geq 27^\circ$, $\leq 39^\circ$ (note that vendor-recommended specific gravity limits are normally more restrictive than those listed here and are normally reflected in site procedures),
 2. a kinematic viscosity at 40 °C of ≥ 1.9 centistokes and ≤ 4.1 centistokes,
 3. a flash point of ≥ 125 °F, and
 4. water and sediment within limits.

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO since the fuel oil is not added to the storage tanks.

Following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-81 (Ref. 4) are met for new fuel oil when tested in accordance with ASTM D975-81 (Ref. 4), except that the analysis for sulfur may be performed in accordance with ASTM D1552-90 (Ref. 4) or ASTM D2622-87 (Ref. 4). These additional analyses are required by Specification 5.5.13, "Diesel Fuel Oil Testing Program," to be performed within 31 days following sampling and addition. This 31 days is intended to assure: 1) that the sample taken is not more than 31 days old at the time of adding the fuel oil to the storage tank, and 2) that the results of a new fuel oil sample (sample obtained prior to addition but not more than 31 days prior to) are obtained within 31 days after addition.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.2 (continued)

For circumstances where multiple fuel oil additions are made within a short period of time, the samples taken for each batch added to the storage tank can be composited for a single follow-up analysis. The 31-day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-88, Method A (Ref. 4). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Each tank is considered and tested separately.

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

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The 31-day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.~~

SR 3.8.3.4

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 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2).~~ 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 performance of the Surveillance.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge helps to ensure the effectiveness of the battery chargers, which support the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state while supplying the continuous steady-state loads of the associated DC subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the minimum float voltage established by the battery manufacturer (2.20 Vpc times the number of connected cells or 127.6 V for a 58 cell battery at the battery terminals). This voltage maintains the battery plates in a condition that supports maintaining the grid life. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The 7-day Frequency is consistent with manufacturer recommendations.~~

SR 3.8.4.2

This SR verifies the design capacity of the chargers. According to Regulatory Guide (RG) 1.32 (Ref. 9), the battery charger supply is recommended 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 ensure that these requirements can be satisfied.

This SR provides two options. One option requires that each battery charger be capable of supplying ≥ 300 amps at the minimum established float voltage for 8 hours. The ampere requirements are based on the output rating of the chargers. The voltage requirements are based on the charger voltage level after a response to a loss of AC power. The time period is sufficient for the charger temperature to have stabilized and to have been maintained for at least 2 hours.

The other option requires that each battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the battery service test and will need to be supplemented with additional loads. The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage, temperature, and the exponential decay in charging current. The battery is recharged when the measured charging current is ≤ 2 amps.

[The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The Surveillance 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 18 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.3

A battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. ~~The Surveillance Frequency of 18 months is consistent with the recommendations of RG 1.32 (Ref. 9) and RG 1.129 (Ref. 10), which state that the battery service test should be performed during refueling operations, or at some other outage, with intervals between tests not to exceed 18 months.~~

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

A modified performance discharge test may be performed in lieu of a service test. The modified performance discharge test (Ref. 8) 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 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 service test for the duration of time equal to that of the service test.

A modified performance discharge test is a test of the battery capacity, as found, 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 and the test discharge rate must envelope the duty cycle of the service test if the modified performance discharge test is performed in lieu of a service test.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

Verifying battery float current while on float charge is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The equipment used to monitor float current must have the necessary accuracy and capability to measure electrical currents in the expected range. The float current requirements are based on the float current indicative of a charged battery. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The 7-day Frequency is consistent with IEEE 450 (Ref. 3).~~

This SR is modified by a Note that states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.4.1. When this float voltage is not maintained the Required Actions of LCO 3.8.4 ACTION A are being taken, which provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limit of 2 amps is established based on the nominal float voltage value and is not directly applicable when this voltage is not maintained.

SR 3.8.6.2 and SR 3.8.6.5

Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to 127.6 V at the battery terminals, or 2.20 Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than 2.07 Vpc, are addressed in Specification 5.5.6. SRs 3.8.6.2 and 3.8.6.5 require verification that the cell float voltages are equal to or greater than the short term absolute minimum voltage of 2.07 V. [The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program](#)~~The Frequency for cell voltage verification every 31 days for pilot cell and 92 days for each connected cell is consistent with IEEE 450 (Ref. 3).~~

SR 3.8.6.3

The limit specified for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability. The minimum design electrolyte level is the minimum level indication mark on the battery cell jar. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The Frequency is consistent with IEEE 450 (Ref. 3).~~

SR 3.8.6.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., 60 °F). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provide the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity. [The Surveillance Frequency is controlled under the Surveillance Frequency Control Program](#)~~The Frequency is consistent with IEEE 450 (Ref. 3).~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.6

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 described in the Bases for SR 3.8.4.3. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.6.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.4.3.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 3), which recommends that the battery be replaced if its capacity is below 80% of the manufacturer's 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. Furthermore, the battery is sized to meet the assumed duty cycle loads when the battery design capacity reaches this 80% limit.

~~The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance 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 Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity $\geq 100\%$ of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 3), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is $\geq 10\%$ below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 3).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. SAR, Chapters 8 and 14.
 2. 10 CFR 50.36.
 3. IEEE-450-1995, "Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications."
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SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and 120 VAC buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the 120 VAC buses. ~~The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 7-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.~~

REFERENCES

1. SAR, Chapter 8.
 2. SAR, Chapter 14.
 3. 10 CFR 50.36.
 4. BAW-2441-A, Revision 2, Risk Informed Justification for LCO End-State Changes, September 2006.
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SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and 120 VAC vital buses energized from the inverter. The verification of proper voltage output ensures that the required power is readily available for the instrumentation connected to the 120 VAC vital buses. ~~The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 7 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.~~

REFERENCES

1. 10 CFR 50.36.
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SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the required AC, DC, and 120 VAC bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained. ~~The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 7-day Frequency takes into account the redundant capability of the AC, DC, and 120 VAC bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.~~

REFERENCES

1. SAR, Chapter 14.
 2. 10 CFR 50.36.
 3. BAW-2441-A, Revision 2, Risk Informed Justification for LCO End-State Changes, September 2006.
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ACTIONS (continued)

A.1, A.2.1, A.2.2, A.2.3, A.2.4, A.2.5, and A.2.6 (continued)

Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that which would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of a fuel handling accident. 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 decay heat removal (DHR) subsystem or a required low temperature overpressure protection (LTOP) feature may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation, heat removal and LTOP. Pursuant to LCO 3.0.6, the DHR ACTIONS and LTOP ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring DHR inoperable, which results in taking the appropriate DHR actions and Required Action A.2.6 is provided to direct entry into the appropriate LTOP Conditions and Required Actions, which results in taking the appropriate LTOP actions.

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 Surveillance verifies that the required AC, DC, and 120 VAC vital bus electrical power distribution subsystems are functioning properly, with all the buses energized. ~~The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 7-day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.~~

ACTIONS (continued)

A.3 (continued)

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS and the refueling canal is within the COLR limits. The boron concentration of the coolant in each volume is determined ~~preiodically every 72 hours~~ by chemical analysis. Prior to re-connecting portions of the refueling canal to the RCS, this SR must be met per SR 3.0.4. If any dilution activity has occurred while the cavity was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS.

~~The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency is based on industry experience, which has shown 72 hours to be adequate.~~

REFERENCES

1. SAR, Section 1.4, GDC 26.
 2. 10 CFR 50.36.
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SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which 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 instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Changes in fuel loading and core geometry can also result in significant differences between source range channels, but each channel should be consistent with its local conditions. When in MODE 6 with only one channel OPERABLE, a CHANNEL CHECK is still required. However, in this condition, a redundant source range instrument may not be available for comparison. The CHANNEL CHECK provides verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for the same instruments in LCO 3.3.9.

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION ~~every 18 months~~. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range nuclear instrument is a complete check and re-adjustment of the channel, from the pre-amplifier input to the indicator. The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18 month Frequency is based on industry experience which has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. SAR, Section 1.4, GDC 13, GDC 26, GDC 28, and GDC 29.
 2. SAR, Section 14.1.2.4.
 3. 10 CFR 50.36.
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APPLICABILITY

The reactor building penetration requirements are applicable during movement of irradiated fuel assemblies within the reactor building because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, the reactor building penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within the reactor building is not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on reactor building penetration status.

ACTIONS

A.1

With the reactor building equipment hatch, air locks, or any reactor building penetration that provides direct access from the reactor building atmosphere to the outside atmosphere not in the required status, 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 reactor building. Performance of this action shall not preclude moving a component to a safe position.

These actions remove the potential for an event which may require reactor building closure to prevent a significant radioactivity release.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the reactor building penetrations required to be in its closed position is in that position.

~~The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Surveillance is performed every 7 days during the movement of irradiated fuel assemblies within the reactor building. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations.~~

This Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the reactor building will not result in a release of fission product radioactivity to the environment in excess of that recommended by Standard Review Plan Section 15.7.4 (Ref. 1, 3 and 6).

SR 3.9.3.2

This Surveillance demonstrates that each reactor building isolation valve actuates to its isolation position on manual initiation. ~~The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18-month Frequency maintains consistency with other similar reactor building isolation valve testing requirements found in Section 3.6.~~ This

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.3.2 (continued)

The SR is modified by a Note stating that this surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

SR 3.9.3.3

This SR requires a CHANNEL CALIBRATION of the reactor building purge exhaust radiation monitor. The CHANNEL CALIBRATION is a complete check of the instrument loop and sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The CHANNEL CALIBRATION is performed consistent with the setpoint requirements. ~~The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 18 month Frequency is based on operating experience and is consistent with the typical operating cycle.~~

REFERENCES

1. Safety Evaluation Report related to ANO-1 Amendment No. 195, April 16, 1999.
 2. SAR, Section 5.2.2.1.3.
 3. Safety Evaluation Report related to ANO-1 Amendment No. 184, September 20, 1996.
 4. SAR, Section 14.2.2.3.
 5. 10 CFR 50.36.
 6. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.
 7. Safety Evaluation Report related to ANO-1 Amendment No. 245, August 10, 2011.
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ACTIONS (continued)

A.2

If DHR loop requirements are not met, actions shall be taken immediately to suspend the loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling canal water level 23 feet above the fuel assemblies seated in the reactor vessel provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading an irradiated fuel assembly, is prudent under this condition.

A.3

If DHR loop requirements are not met, actions shall be initiated immediately in order to satisfy DHR loop requirements.

Restoration of one decay heat removal loop is required because this is the only active method of removing decay heat. Dissipation of decay heat through natural convection to the large inventory of water in the refueling canal should not be relied upon for an extended period of time. The immediate Completion Time reflects the importance of restoring an adequate decay heat removal loop.

A.4

If DHR loop requirements are not met, all reactor building penetrations providing direct access from the reactor building atmosphere to outside atmosphere shall be closed within 4 hours.

If no means of decay heat removal can be restored, the core decay heat could raise temperatures and cause boiling in the core which could result in increased levels of radioactivity in the reactor building atmosphere. Closure of the penetrations providing access to the outside atmosphere will prevent the uncontrolled release of radioactivity to the environment.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the DHR loop is in operation and circulating reactor coolant. Verification includes flow, temperature, or pump status monitoring, which help assure that forced flow is providing heat removal. ~~The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 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 DHR System.~~

ACTIONS (continued)

B.1

If no DHR loop is in operation or no DHR loop is OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

B.2

If no DHR loop is in operation or no DHR loop is OPERABLE, actions shall be initiated immediately and continued without interruption to restore one DHR loop to OPERABLE status and operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE DHR loops and one operating DHR loop should be accomplished expeditiously.

If no DHR loop is OPERABLE or in operation, alternate actions shall have been initiated immediately under Condition A to establish ≥ 23 ft of water above the top of fuel assemblies seated in the reactor vessel. Furthermore, when the LCO cannot be fulfilled, alternate decay heat removal methods, as specified in the unit's Abnormal and Emergency Operating Procedures, should be implemented. The method used to remove decay heat should be the most prudent as well as the safest choice, based upon unit conditions. The choice could be different if the reactor vessel head is in place rather than removed.

B.3

If no DHR loop is in operation, all reactor building penetrations providing direct access from the reactor building atmosphere to the outside atmosphere must be closed within 4 hours. With the DHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the reactor building atmosphere. Closing reactor building penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one DHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help assure that forced flow is providing heat removal.

The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the DHR system in the control room.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.5.2

Verification that each required pump is available ensures that an additional DHR pump 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. Alternatively, verification that a DHR pump is in operation as required by SR 3.9.4.1 also verifies proper breaker alignment and power availability. ~~The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

REFERENCES

1. SAR, Section 1.4.
 2. SAR, Section 9.5.
 3. 10 CFR 50.36.
-

APPLICABILITY

LCO 3.9.6 is applicable during movement of irradiated fuel assemblies within the reactor building. The LCO minimizes the possibility of a fuel handling accident in the reactor building that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in the reactor building, there can be no significant radioactivity release as a result of a postulated fuel handling accident in the reactor building.

ACTIONS

A.1

With a water level of < 23 feet above the top of the irradiated fuel assemblies seated with the reactor pressure vessel, all operations involving the movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of irradiated fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated within the reactor pressure 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 within the reactor pressure vessel limits the consequences of damaged fuel rods that are postulated to result from a postulated fuel handling accident inside the reactor building (Ref. 2).

The periodic Surveillance Frequency is controlled under the Surveillance Frequency Control Program. 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, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.183.
 2. SAR Section 14.2.2.3.
 3. 10 CFR 50.67.
 4. 10 CFR 50.36.
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ATTACHMENT 6

1CAN031801

PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATIONS

PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATIONS

Description of Amendment Request:

The change requests the adoption of an approved change to the standard technical specifications (STS) for Babcock and Wilcox Plants (NUREG-1430), to allow relocation of specific Arkansas Nuclear One – Unit 1 (ANO-1) TS surveillance frequencies to a licensee-controlled program. The proposed change is described in Technical Specification Task Force (TSTF) Traveler, TSTF-425, Revision 3 (Rev. 3) (ADAMS Accession No. ML090850642) related to the Relocation of Surveillance Frequencies to Licensee Control – Risk Informed TSTF (RITSTF) Initiative 5b, and was described in the Notice of Availability published in the Federal Register on July 6, 2009 (74 FR 31996).

The proposed changes are consistent with NRC-approved Industry/Technical Specification Task Force (TSTF) Traveler, TSTF-425, Rev. 3, “Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b.” The proposed change relocates surveillance frequencies to a licensee-controlled program, the Surveillance Frequency Control Program (SFPC). 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).

Basis for proposed no significant hazards consideration:

As required by 10 CFR 50.91(a), Entergy Operations Inc. (Entergy's) analysis of the issue of no significant hazards consideration is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

Response: No.

The proposed change relocates the specified frequencies for periodic surveillance requirements (SRs) to licensee control under a new Surveillance Frequency Control Program (SFPC). 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 (TSs) for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the SRs, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

2. Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

Response: No.

No new or different accidents result from utilizing the proposed change. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in the margin of safety?

Response: No.

The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to TS), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, Entergy will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Rev. 1 in accordance with the TS SFCP. NEI 04-10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above, Entergy concludes that the requested change does not involve a significant hazards consideration as set forth in 10 CFR 50.92(c), Issuance of amendment.

ATTACHMENT 7

1CAN031801

ANO-1 TO NUREG 1430 SR CROSS-REFERENCE

Legend:

Arkansas Nuclear One, Unit 1 (ANO-1) Surveillance Requirement (SR) Frequency Identified for Relocation Not Included in TSTF-425 = Gray Row

ANO-1 SR #	NUREG-1430 R4 SR #	ANO-1 Surveillance Description {NUREG 1430 Description if no ANO-1 Surveillance}	ANO-1 Surveillance Frequency	SR Frequency Modified by TSTF-425	SR Frequency Modified in Proposed Amendment Request
Section 1.0, Definitions					
Definition	Definitions	Staggered Test Basis	N/A	yes	yes
Section 3.1, Reactivity Control Systems					
3.1.1.1	3.1.1.1	Verify SDM [Shutdown Margin] greater than or equal to the limit specified in the COLR.	24 hours	yes	yes
3.1.2.1	3.1.2.1	Verify measured core reactivity balance is within $\pm 1\%$ $\Delta k/k$ of predicted values.	Once prior to entering MODE 1 after each fuel loading AND 31 EFPD thereafter	yes (31-EFPD Frequency only)	yes (31-EFPD Frequency only)
3.1.4.1	3.1.4.1	Verify individual CONTROL ROD positions are within 6.5% of their group average height.	12 hours	yes	yes
3.1.4.2	3.1.4.2	Verify CONTROL ROD freedom of movement for each individual CONTROL ROD that is not fully inserted.	92 days	yes	yes
3.1.5.1	3.1.5.1	Verify each safety rod is fully withdrawn	12 hours	yes	yes
3.1.6.1	3.1.6.1	Verify position of each APSR [Axial Power Shaping Rod] is within 6.5% of the group average height.	12 hours	yes	yes
3.1.7.1	3.1.7.1	Perform CHANNEL CHECK of required position indicator channel	12 hours	yes	yes
3.1.7.2	N/A	Perform CHANNEL CALIBRATION of required position indicator channel.	18 months	no	yes
3.1.8.1	3.1.8.1	Verify THERMAL POWER is $\leq 85\%$ RTP	1 hour	yes	yes
3.1.8.2	3.1.8.2	Perform SR 3.2.5.1.	2 hours	yes	yes

ANO-1 SR #	NUREG-1430 R4 SR #	ANO-1 Surveillance Description {NUREG 1430 Description if no ANO-1 Surveillance}	ANO-1 Surveillance Frequency	SR Frequency Modified by TSTF-425	SR Frequency Modified in Proposed Amendment Request
3.1.8.3	3.1.8.3	Verify nuclear overpower trip setpoint \leq 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP.	Within 8 hours prior to performance of PHYSICS TESTS at each test plateau	yes	no
3.1.8.4	3.1.8.4	Verify SDM to be within the limits provided in the COLR.	24 hours	yes	yes
3.1.9.1	3.1.9.1	Verify THERMAL POWER is \leq 5% RTP	1 hour	yes	yes
3.1.9.2	3.1.9.2	Verify nuclear overpower trip setpoint is \leq 5% RTP.	Within 8 hours prior to performance of PHYSICS TESTS	yes	no
3.1.9.3	3.1.9.3	Verify SDM to be within the limit provided in the COLR.	24 hours	yes	yes
Section 3.2, Power Distribution Limits					
3.2.1.1	3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	12 hours	yes	yes
3.2.1.2	3.2.1.2	Verify regulating rod groups meet the insertion limits as specified in the COLR.	12 hours	yes	yes
3.2.2.1	3.2.2.1	Verify APSRs are within acceptable limits specified in the COLR.	12 hours	yes	yes
3.2.3.1	3.2.3.1	Verify AXIAL POWER IMBALANCE is within limits as specified in the COLR.	12 hours	yes	yes
3.2.4.1	3.2.4.1	Verify QPT is within limits as specified in the COLR	7 days AND When QPT has been restored to less than or equal to the steady state limit, 1 hour for 12 consecutive hours, or until verified acceptable at \geq 95% RTP	yes (7-day Frequency only)	yes (7-day Frequency only)

ANO-1 SR #	NUREG-1430 R4 SR #	ANO-1 Surveillance Description {NUREG 1430 Description if no ANO-1 Surveillance}	ANO-1 Surveillance Frequency	SR Frequency Modified by TSTF-425	SR Frequency Modified in Proposed Amendment Request
Section 3.3, Instrumentation					
3.3.1.1	3.3.1.1	Perform CHANNEL CHECK	12 hours	yes	yes
3.3.1.2	3.3.1.2	Compare results of calorimetric heat balance calculation to power range channel output	96 hours AND Once within 24 hours after a THERMAL POWER change of ≥ 10% RTP	yes (96-hour Frequency only)	yes (96-hour Frequency only)
3.3.1.3	3.3.1.3	Compare results of out of core measured AXIAL POWER IMBALANCE to incore measured AXIAL POWER IMBALANCE.	31 days	yes	yes
3.3.1.4	3.3.1.4	Perform CHANNEL FUNCTIONAL TEST	31 days	yes	yes
3.3.1.5	3.3.1.5	Perform CHANNEL CALIBRATION.	18 months	yes	yes
N/A	3.3.1.6	{Verify that RPS RESPONSE TIME is within limits}	N/A	yes	no
3.3.3.1	3.3.3.1	Perform CHANNEL FUNCTIONAL TEST	92 days	yes	yes
3.3.4.1	3.3.4.1	Perform CHANNEL FUNCTIONAL TEST	92 days	yes	yes
3.3.5.1	3.3.5.1	Perform CHANNEL CHECK	12 hours	yes	yes
3.3.5.2	3.3.5.2	Perform CHANNEL FUNCTIONAL TEST	31 days	yes	yes
3.3.5.3	3.3.5.3	Perform CHANNEL CALIBRATION	18 months	yes	yes
N/A	3.3.5.4	{Verify ESFAS RESPONSE TIME within limits}	N/A	yes	no
3.3.6.1	3.3.6.1	Perform CHANNEL FUNCTIONAL TEST	18 months	yes	yes
3.3.7.1	3.3.7.1	Perform digital actuation logic CHANNEL FUNCTIONAL TEST.	31 days	yes	yes
3.3.8.1	3.3.8.1	Perform CHANNEL CHECK	7 days	yes	yes
N/A	3.3.8.2	{Perform CHANNEL FUNCTIONAL TEST}	N/A	yes	no

ANO-1 SR #	NUREG-1430 R4 SR #	ANO-1 Surveillance Description {NUREG 1430 Description if no ANO-1 Surveillance}	ANO-1 Surveillance Frequency	SR Frequency Modified by TSTF-425	SR Frequency Modified in Proposed Amendment Request
3.3.8.2	3.3.8.3	Perform CHANNEL CALIBRATION with setpoint Allowable Value as follows: a. Degraded voltage ≥ 423.2 V and ≤ 436.0 V with a time delay of 8 seconds ± 1 second; and b. Loss of voltage ≥ 1600 V and ≤ 3000 V with a time delay of ≥ 0.30 seconds and ≤ 0.98 seconds.	18 months	yes	yes
3.3.9.1	3.3.9.1	Perform CHANNEL CHECK	12 hours	yes	yes
3.3.9.2	3.3.9.2	Perform CHANNEL CALIBRATION	18 months	yes	yes
3.3.10.1	3.3.10.1	Perform CHANNEL CHECK	12 hours	yes	yes
3.3.10.2	N/A	Perform CHANNEL FUNCTIONAL TEST	31 days	no	yes
3.3.10.3	3.3.10.2	Perform CHANNEL CALIBRATION	18 months	yes	yes
3.3.11.1	3.3.11.1	Perform CHANNEL CHECK	12 hours	yes	yes
3.3.11.2	3.3.11.2	Perform CHANNEL FUNCTIONAL TEST	31 days	yes	yes
3.3.11.3	3.3.11.3	Perform CHANNEL CALIBRATION	18 months	yes	yes
N/A	3.3.11.4	{Verify EFIC RESPONSE TIME is within limits}	N/A	yes	no
3.3.12.1	3.3.12.1	Perform CHANNEL FUNCTIONAL TEST	31 days	yes	yes
3.3.13.1	3.3.13.1	Perform CHANNEL FUNCTIONAL TEST	31 days	yes	yes
3.3.14.1	3.3.14.1	Perform CHANNEL FUNCTIONAL TEST	31 days	yes	yes
N/A	3.3.15.1	{Perform CHANNEL CHECK}	N/A	yes	no
N/A	3.3.15.2	{Perform CHANNEL FUNCTIONAL TEST}	N/A	yes	no
N/A	3.3.15.3	{Perform CHANNEL CALIBRATION with setpoint Allowable Value $\leq [25]$ mR/hr}	N/A	yes	no
3.3.15.1	3.3.17.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days	yes	yes
3.3.15.2	3.3.17.2	Perform CHANNEL CALIBRATION.	18 months	yes	yes
3.3.16.1	3.3.16.1	Perform CHANNEL CHECK	12 hours	yes	yes

ANO-1 SR #	NUREG-1430 R4 SR #	ANO-1 Surveillance Description {NUREG 1430 Description if no ANO-1 Surveillance}	ANO-1 Surveillance Frequency	SR Frequency Modified by TSTF-425	SR Frequency Modified in Proposed Amendment Request
3.3.16.2	3.3.16.2	Perform CHANNEL FUNCTIONAL TEST	31 days	yes	yes
3.3.16.3	3.3.16.3	Perform CHANNEL CALIBRATION	18 months	yes	yes
N/A	3.3.18.1	{[Perform CHANNEL CHECK for each required instrumentation channel that is normally energized]}	N/A	yes	no
N/A	3.3.18.2	{Verify each required control circuit and transfer switch is capable of performing the intended function.}	N/A	yes	no
N/A	3.3.18.3	{Perform CHANNEL CALIBRATION for each required instrumentation channel.}	N/A	yes	no
Section 3.4, Reactor Coolant System					
3.4.1.1	3.4.1.1	Verify RCS loop pressure is within the limit specified in the COLR.	12 hours	yes	yes
3.4.1.2	3.4.1.2	Verify RCS hot leg temperature is within the limit specified in the COLR.	12 hours	yes	yes
3.4.1.3	3.4.1.3	Verify RCS total flow is within the limit specified in the COLR.	12 hours	yes	yes
3.4.1.4	3.4.1.4	Verify RCS total flow rate is within the limit specified in the COLR by measurement.	18 months	yes	yes
3.4.2.1	3.4.2.1	Verify RCS Tavg ≥ 525 °F.	12 hours	yes	yes
3.4.3.1	3.4.3.1	Verify RCS pressure, RCS temperature, and RCS heatup rates are within the limits specified in Figure 3.4.3-1.	30 minutes	yes	yes
3.4.3.2	N/A	Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-2.	30 minutes	no	yes
3.4.3.3	N/A	Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-3.	30 minutes	no	yes
3.4.3.4	N/A	Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.3-1.	30 minutes	no	yes
3.4.4.1	3.4.4.1	Verify required RCS loops are in operation.	12 hours	yes	yes
3.4.5.1	3.4.5.1	Verify required RCS loop is in operation.	12 hours	yes	yes

ANO-1 SR #	NUREG-1430 R4 SR #	ANO-1 Surveillance Description {NUREG 1430 Description if no ANO-1 Surveillance}	ANO-1 Surveillance Frequency	SR Frequency Modified by TSTF-425	SR Frequency Modified in Proposed Amendment Request
3.4.5.2	3.4.5.2	Verify correct breaker alignment and indicated power available to each required pump.	7 days	yes	yes
3.4.6.1	3.4.6.1	Verify required DHR or RCS loop is in operation	12 hours	yes	yes
3.4.6.2	3.4.6.2	Verify correct breaker alignment and indicated power available to each required pump	7 days	yes	yes
3.4.7.1	3.4.7.1	Verify required DHR loop is in operation	12 hours	yes	yes
3.4.7.2	3.4.7.2	Verify required SG secondary side water levels are ≥ 20 inches.	12 hours	yes	yes
3.4.7.3	3.4.7.3	Verify correct breaker alignment and indicated power available to each required DHR pump.	7 days	yes	yes
3.4.8.1	3.4.8.1	Verify required DHR loop is in operation	12 hours	yes	yes
3.4.8.2	3.4.8.2	Verify correct breaker alignment and indicated power available to each required DHR pump.	7 days	yes	yes
3.4.9.1	3.4.9.1	Verify pressurizer water level ≤ 320 inches.	12 hours	yes	yes
3.4.9.2	3.4.9.2	Verify capacity of ES bus powered pressurizer heaters ≥ 126 kW.	18 months	yes	yes
N/A	3.4.9.3	{[Verify emergency power supply for pressurizer heaters is OPERABLE]}	N/A	yes	no
N/A	3.4.11.1	{Perform one complete cycle of the block valve}	N/A	yes	no
N/A	3.4.11.2	{Perform one complete cycle of the PORV}	N/A	yes	no
N/A	3.4.11.3	{[Verify PORV and block valve are capable of being powered from an emergency power source.]}	N/A	yes	no
N/A	3.4.12.1	{Verify a maximum of [one] makeup pump is capable of injecting into the RCS}	N/A	yes	no
3.4.11.1	3.4.12.4	Verify pressurizer level does not represent a water solid condition	30 minutes during RCS heatup and cooldown AND 12 hours	yes (12-hour Frequency only)	yes (12-hour Frequency only)
3.4.11.2	3.4.12.2	Verify HPI is deactivated	12 hours	yes	yes

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3.4.11.3	3.4.12.3	Verify each pressurized CFT is isolated.	12 hours	yes	yes
3.4.11.4	3.4.12.5 3.4.12.6	Verify OPERABLE pressure relief capability	12 hours	yes	yes
N/A	3.4.12.7	{Perform CHANNEL FUNCTIONAL TEST for PORV}	N/A	yes	no
3.4.11.5	3.4.12.8	Perform CHANNEL CALIBRATION of ERV opening circuitry.	18 months	yes	yes
3.4.12.1	3.4.16.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity $\leq 2200 \mu\text{Ci/gm}$.	7 days	yes	yes
3.4.12.2	3.4.16.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq 1.0 \mu\text{Ci/gm}$.	14 days	yes	yes
N/A	3.4.16.3	{Determine E bar}	N/A	yes	no
3.4.13.1	3.4.13.1	Verify RCS operational LEAKAGE is within limits by performance of an RCS water inventory balance.	72 hours	yes	yes
3.4.13.2	3.4.13.2	Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	72 hours	yes	yes
3.4.14.1	3.4.14.1	Verify leakage from each RCS pressure isolation check valve, or pair of check valves, as applicable, is less than or equal to an equivalent of the Allowable Leakage Limit identified below at a differential test pressure ≥ 150 psid.	INSERVICE TESTING PROGRAM AND Once prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months	yes	no
3.4.14.2	3.4.14.2	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual high RCS pressure signal.	18 months	yes	yes

ANO-1 SR #	NUREG-1430 R4 SR #	ANO-1 Surveillance Description {NUREG 1430 Description if no ANO-1 Surveillance}	ANO-1 Surveillance Frequency	SR Frequency Modified by TSTF-425	SR Frequency Modified in Proposed Amendment Request
3.4.14.3	3.4.14.3	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual high RCS pressure signal: a. ≤ 340 psig for one valve; and b. ≤ 400 psig for the other valve.	18 months	yes	yes
3.4.14.4	N/A	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual Core Flood Tank isolation valve "not closed" signal.	18 months	no	yes
3.4.14.5	N/A	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual Core Flood Tank isolation valve "not closed" signal.	18 months	no	yes
3.4.15.1	3.4.15.1	Perform CHANNEL CHECK of required reactor building atmosphere radioactivity monitor.	12 hours	yes	yes
3.4.15.2	3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of required reactor building atmosphere radioactivity monitor	92 days	yes	yes
3.4.15.3	3.4.15.4	Perform CHANNEL CALIBRATION of required reactor building atmosphere radioactivity monitor.	18 months	yes	yes
3.4.15.4	3.4.15.3	Perform CHANNEL CALIBRATION of required reactor building sump monitor.	18 months	yes	yes
Section 3.5, Emergency Core Cooling Systems (ECCS)					
3.5.1.1	3.5.1.1	Verify each CFT isolation valve is fully open	12 hours	yes	yes
3.5.1.2	3.5.1.2	Verify borated water volume in each CFT is ≥ 970 ft ³ and ≤ 1110 ft ³ .	12 hours	yes	yes
3.5.1.3	3.5.1.3	Verify nitrogen cover pressure in each CFT is ≥ 560 psig and ≤ 640 psig.	12 hours	yes	yes

ANO-1 SR #	NUREG-1430 R4 SR #	ANO-1 Surveillance Description {NUREG 1430 Description if no ANO-1 Surveillance}	ANO-1 Surveillance Frequency	SR Frequency Modified by TSTF-425	SR Frequency Modified in Proposed Amendment Request
3.5.1.4	3.5.1.4	Verify boron concentration in each CFT is ≥ 2270 ppm.	31 days AND Once within 12 hours after each solution level increase of ≥ 0.2 feet that is not the result of addition from a borated water source of known concentration ≥ 2270 ppm	yes (31-day Frequency only)	yes (31-day Frequency only)
3.5.1.5	3.5.1.5	Verify power is removed from each CFT isolation valve operator.	31 days	yes	yes
N/A	3.5.2.1	{[Verify the following valves are in the listed position with power to the valve operator removed.]}	N/A	yes	no
3.5.2.1	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	yes	yes
N/A	3.5.2.3	{[Verify ECCS piping is full of water.]}	N/A	yes	no
3.5.2.3	3.5.2.5	Verify each ECCS 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.	18 months	yes	yes
3.5.2.4	3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months	yes	yes
N/A	3.5.2.7	{[Verify the correct settings of stops for the following HPI stop check valves:]}	N/A	yes	no
N/A	3.5.2.8	{[Verify the flow controllers for the following LPI throttle valves operate properly:]}	N/A	yes	no
3.5.2.5	3.5.2.9	Verify, by visual inspection, each ECCS train reactor building sump suction inlet is not restricted by debris and screens show no evidence of structural distress or abnormal corrosion.	18 months	yes	yes
3.5.4.1	3.5.4.1	Verify BWST borated water temperature is ≥ 40 °F and ≤ 110 °F.	24 hours	yes	yes

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3.5.4.2	3.5.4.2	Verify BWST borated water level is ≥ 38.4 feet and ≤ 42 feet.	7 days	yes	yes
3.5.4.3	3.5.4.3	Verify BWST boron concentration ≥ 2270 ppm and ≤ 2670 ppm.	7 days	yes	yes
Section 3.6, Reactor Building Systems					
3.6.2.2	3.6.2.2	Verify only one door in the air lock can be opened at a time.	18 months	yes	yes
3.6.3.1	3.6.3.1 3.6.3.2	Verify each reactor building purge isolation valve is closed.	31 days	yes	yes
3.6.3.2	3.6.3.3	Verify each reactor building isolation manual valve and blind flange that is located outside the reactor building and not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	31 days	yes	yes
3.6.3.4	3.6.3.5	Verify the isolation time of each automatic power operated reactor building isolation valve is within limits.	INSERVICE TESTING PROGRAM	yes	no
N/A	3.6.3.6	{Perform leakage rate testing for containment purge valves with resilient seals}	N/A	yes	no
3.6.3.5	3.6.3.7	Verify each automatic reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months	yes	yes
N/A	3.6.3.8	{[Verify each [] inch containment purge valve is blocked to restrict the valve from opening > [50]%.]}	N/A	yes	no
3.6.4.1	3.6.4.1	Verify reactor building pressure is ≥ -1.0 psig and $\leq +3.0$ psig.	12 hours	yes	yes
N/A	3.6.5.1	{Verify containment average air temperature is within limit.}	N/A	yes	no
3.6.5.1	3.6.6.1	Verify each reactor building spray manual, power operated, and automatic valve in each required flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days	yes	yes
3.6.5.2	3.6.6.2	Operate each required reactor building cooling train fan unit for ≥ 15 minutes.	31 days	yes	yes

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3.6.5.3	3.6.6.3	Verify each required reactor building cooling train cooling water flow rate is ≥ 1200 gpm.	31 days	yes	yes
3.6.5.5	3.6.6.5	Verify each automatic reactor building spray valve in each required flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months	yes	yes
3.6.5.6	3.6.6.6	Verify each required reactor building spray pump starts automatically on an actual or simulated actuation signal.	18 months	yes	yes
3.6.5.7	3.6.6.7	Verify each required reactor building cooling train starts automatically on an actual or simulated actuation signal.	18 months	yes	yes
3.6.5.8	3.6.6.8	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage	yes	no
3.6.6.1	3.6.7.1	Verify each Spray Additive System 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	yes	yes
3.6.6.2	3.6.7.2	Verify sodium hydroxide tank solution volume is ≥ 9000 gallons.	184 days	yes	yes
3.6.6.3	3.6.7.3	Verify sodium hydroxide tank solution concentration is > 6.0 wt% and < 8.5 wt.% NaOH.	184 days	yes	yes
3.6.6.4	3.6.7.4	Verify each Spray Additive System automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	18 months	yes	yes
N/A	3.6.7.5	{Verify Spray Additive System flow [rate] from each solution's flow path.}	N/A	yes	no
Section 3.7, Plant Systems					
3.7.2.2	3.7.2.2	Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	18 months	yes	yes

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3.7.3.2	3.7.3.2	Verify that each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve actuates to the isolation position on an actual or simulated actuation signal.	18 months	yes	yes
N/A	3.7.4.1	{Verify one complete cycle of each AVV [Atmospheric Vent Valve].}	N/A	yes	no
N/A	3.7.4.2	{Verify one complete cycle of each AVV block valve.}	N/A	yes	no
3.7.4.1	3.7.17.1	Verify the specific activity of the secondary coolant is $\leq 0.1 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.	31 days	yes	yes
3.7.5.1	3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days	yes	yes
3.7.5.3	3.7.5.3	Verify each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months	yes	yes
3.7.5.4	3.7.5.4	Verify each EFW pump starts automatically on an actual or simulated actuation signal.	18 months	yes	yes
3.7.5.6	N/A	Verify that feedwater is delivered to each steam generator using the motor-driven EFW pump.	18 months	no	yes
N/A	3.7.5.6	{[Perform a CHANNEL FUNCTIONAL TEST for the EFW pump suction pressure interlocks.]}	N/A	yes	no
N/A	3.7.5.7	{[Perform a CHANNEL CALIBRATION for the EFW pump suction pressure interlocks.]}	N/A	yes	no
3.7.6.1	3.7.6.1	Verify QCST volume is $\geq 267,000$ gallons when required for both units and $\geq 107,000$ gallons when only required for Unit 1.	12 hours	yes	yes
N/A	3.7.7.1	{Verify each CCW 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.}	N/A	yes	no

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N/A	3.7.7.2	{Verify each CCW 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.}	N/A	yes	no
N/A	3.7.7.3	{Verify each CCW pump starts automatically on an actual or simulated actuation signal.}	N/A	yes	no
3.7.7.1	3.7.8.1	Verify each SWS 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	yes	yes
3.7.7.2	3.7.8.2	Verify each SWS 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.	18 months	yes	yes
3.7.7.3	3.7.8.3	Verify each required SWS pump starts automatically on an actual or simulated signal.	18 months	yes	yes
3.7.8.1	3.7.9.1	Verify that the indicated water level of the ECP [Emergency Cooling Pond] is greater than or equal to that required for an ECP volume of 70 acre-ft.	24 hours	yes	yes
3.7.8.2	3.7.9.2	Verify average water temperature is ≤ 100 °F.	24 hours	yes	yes
N/A	3.7.9.3	{[Operate each cooling tower fan for > [15] minutes.]}	N/A	yes	no
3.7.8.3	N/A	Perform soundings of the ECP to verify: 1. A contained water volume of ECP ≥ 70 acre-feet, and 2. The minimum indicated water level needed to ensure a volume of 70 acre-feet is maintained.	12 months	no	yes
3.7.8.4	N/A	Perform visual inspection of the ECP to verify conformance with design requirements.	12 months	no	yes
3.7.9.1	3.7.10.1	Operate each CREVS [Control Room Emergency Ventilation System] train for ≥ 15 minutes	31 days	yes	yes
3.7.9.3	3.7.10.3	Verify the CREVS automatically isolates the Control Room and switches into a recirculation mode of operation on an actual or simulated actuation signal	18 months	yes	yes

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N/A	3.7.10.5	[Verify the system makeup flow rate is \geq [270] and \leq [330] cfm when supplying the the control room with outside air.]]	N/A	yes	no
3.7.10.1	3.7.11.1	Verify each CREACS [Control Room Emergency Air Conditioning System] train starts, operates for at least 1 hour, and maintains control room air temperature \leq 84 °F D. B.	31 days	yes	yes
3.7.10.2	N/A	Verify system flow rate of 9900 cfm \pm 10%.	18 months	no	yes
3.7.11.1	3.7.12.1	Operate each PRVS [Penetration Room Ventilation System] train for \geq 15 minutes.	31 days	yes	yes
3.7.11.3	3.7.12.3	Verify each PRVS train actuates on an actual or simulated actuation signal.	18 months	yes	yes
N/A	3.7.12.4	{Verify one EVS train can maintain a pressure \leq [] inches water gauge relative to atmospheric pressure during the [post accident] mode of operation at a flow rate of \leq [3000] cfm.}	N/A	yes	no
N/A	3.7.12.5	{[Verify each EVS filter cooling bypass damper can be opened.]}	N/A	yes	no
N/A	3.7.13.1	{[Operate each FSPVS train for \geq 10 continuous hours with the heaters operating or (for systems without heaters) \geq 15 minutes].}	N/A	yes	no
N/A	3.7.13.2	{[Perform required FSPVS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].}	N/A	yes	no
N/A	3.7.13.3	{[Verify each FSPVS train actuates on an actual or simulated actuation signal.]}	N/A	yes	no
N/A	3.7.13.4	{Verify one FSPVS train can maintain a pressure \leq [] inches water gauge with respect to atmospheric pressure during the [post accident] mode of operation at a flow rate \leq [3000] cfm.}	N/A	yes	no
N/A	3.7.13.5	{[Verify each FSPVS filter bypass damper can be opened.]}	N/A	yes	no
3.7.13.1	3.7.14.1	Verify the spent fuel pool water level is \geq 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	7 days	yes	yes
3.7.14.1	3.7.15.1	Verify the spent fuel pool boron concentration is > 2000 ppm.	7 days	yes	yes
N/A	3.7.18.1	{Verify steam generator water level to be within limits.}	N/A	yes	no

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Section 3.8, Electrical Power Systems					
3.8.1.1	3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit	7 days	yes	yes
3.8.1.2	3.8.1.2 3.8.1.7	Verify each DG starts from standby conditions and, in ≤ 15 seconds achieves “ready-to-load” conditions	31 days	yes	yes
3.8.1.3	3.8.1.3	Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2475 kW and ≤ 2750 kW.	31 days	yes	yes
3.8.1.4	3.8.1.4	Verify each day tank contains ≥ 160 gallons of fuel oil	31 days	yes	yes
3.8.1.5	3.8.1.5	Check for and remove accumulated water from each day tank.	31 days	yes	yes
3.8.1.6	3.8.1.6	Verify the fuel oil transfer system operates to transfer fuel oil from storage tanks to the day tank.	31 days	yes	yes
3.8.1.7	3.8.1.8	Verify automatic transfer of AC power sources to the selected offsite circuit and manual transfer to the alternate required offsite circuit.	18 months	yes	yes
N/A	3.8.1.9	{Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ [63] Hz, b. Within [3] seconds following load rejection, the voltage is ≥ [3740] V and ≤ [4580] V, and c. Within [3] seconds following load rejection, the frequency is ≥ [58.8] Hz and ≤ [61.2] Hz.}	N/A	yes	no
N/A	3.8.1.10	{Verify each DG does not trip, and voltage is maintained ≤ [5000] V during and following a load rejection of ≥ [4500] kW and ≤ [5000] kW.}	N/A	yes	no

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3.8.1.8	3.8.1.11	Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. achieves "ready-to-load" conditions in ≤ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected shutdown load through automatic load sequencing timers, and 4. supplies connected loads for ≥ 5 minutes.	18 months	yes	yes
N/A	3.8.1.12	{Verify on an actual or simulated [Engineered Safety Feature (ESF)] actuation signal each DG auto-starts from standby condition and: a. In $\leq [12]$ seconds after auto-start and during tests, achieves voltage $\geq [37\ 40]$ V and frequency $\geq [58.8]$ Hz, b. Achieves steady state voltage $\geq [37\ 40]$ V and $\leq [4580]$ V and frequency $\geq [58.8]$ Hz and $\leq [61.2]$ Hz, c. Operates for ≥ 5 minutes, d. Permanently connected loads remain energized from the offsite power system, and e. Emergency loads are energized [or autoconnected through the automatic load sequencer] from the offsite power system.}	N/A	yes	no
N/A	3.8.1.13	{Verify each DG's noncritical automatic trips are bypassed on [actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal].}	N/A	yes	no
N/A	3.8.1.14	{Verify each DG operates for ≥ 24 hours: a. For $\geq [2]$ hours loaded $\geq [5250]$ kW and $\leq [6000]$ kW and b. For the remaining hours of the test loaded $\geq [4500]$ kW and $\leq [5000]$ kW.}	N/A	yes	no

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N/A	3.8.1.15	{Verify each DG starts and achieves: a. In $\leq [10]$ seconds, voltage $\geq [3740]$ V and frequency $\geq [58.8]$ Hz and b. Steady state voltage $\geq [3740]$ V and $\leq [4580]$ V, and frequency $\geq [58.8]$ Hz and $\leq [61.2]$ Hz.}	N/A	yes	no
N/A	3.8.1.16	{Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power, b. Transfers loads to offsite power source, and c. Returns to ready-to-load operation.}	N/A	yes	no
N/A	3.8.1.17	{Verify, with a DG operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by: a. Returning DG to ready-to-load operation and [b. Automatically energizing the emergency load from offsite power.]}	N/A	yes	no
N/A	3.8.1.18	{Verify interval between each sequenced load block is within $\pm [10\%$ of design interval] for each emergency [and shutdown] load sequencer}	N/A	yes	no
3.8.1.9	3.8.1.19	Verify on an actual or simulated loss of offsite power in conjunction with an actual or simulated ESF actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. achieves "ready-to-load" conditions in ≤ 15 seconds, 2. energizes permanently connected loads, 3. energizes auto-connected shutdown load through automatic load sequencing timers, and 4. supplies connected loads for ≥ 5 minutes.	18 months	yes	yes

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N/A	3.8.1.20	{Verify, when started simultaneously from standby condition, each DG achieves, in $\leq [10]$ seconds, voltage $\geq [3740]$ V and $\leq [4580]$ V, and frequency $\geq [58.8]$ Hz and $\leq [61.2]$ Hz.}	N/A	yes	no
3.8.2.1	3.8.2.1	For AC Sources required to be OPERABLE, the SRs of Specification 3.8.1, "AC Sources – Operating," except SR 3.8.1.4, SR 3.8.1.7, SR 3.8.1.8, and SR 3.8.1.9, are applicable.	31 days	no	yes
3.8.3.1	3.8.3.1	Verify each fuel oil storage tank contains $\geq 20,000$ gallons of fuel.	31 days	yes	yes
N/A	3.8.3.2	{Verify lube oil inventory is \geq a [7] day supply}	N/A	yes	no
3.8.3.3	3.8.3.4	Verify each DG required air start receiver pressure is ≥ 175 psig.	31 days	yes	yes
3.8.3.4	3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	31 days	yes	yes
3.8.4.1	3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	7 days	yes	yes
3.8.4.2	3.8.4.2	Verify each battery charger supplies ≥ 300 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours. OR Verify each battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state	18 months	yes	yes
3.8.4.3	3.8.4.3	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 or a modified performance discharge test	18 months	yes	yes
3.8.6.1	3.8.6.1	Verify each battery float current is ≤ 2 amps	7 days	yes	yes
3.8.6.2	3.8.6.2	Verify each battery pilot cell float voltage is ≥ 2.07 V	31 days	yes	yes
3.8.6.3	3.8.6.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	31 days	yes	yes

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3.8.6.4	3.8.6.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits	31 days	yes	yes
3.8.6.5	3.8.6.5	Verify each battery connected cell float voltage is ≥ 2.07 V.	92 days	yes	yes
3.8.6.6	3.8.6.6	Verify battery capacity is $\geq 80\%$ of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test	60 months 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 $\geq 100\%$ of manufacturer's rating	yes (60-month Frequency only)	yes (60-month Frequency only)
3.8.7.1	3.8.7.1	Verify correct inverter voltage, frequency, and alignment to associated 120 VAC buses RS1, RS2, RS3, and RS4.	7 days	yes	yes
3.8.8.1	3.8.8.1	Verify correct inverter voltage and alignments to required 120 VAC vital buses.	7 days	yes	yes
3.8.9.1	3.8.9.1	Verify correct breaker alignments to required AC, DC, and 120 VAC bus electrical power distribution subsystems.	7 days	yes	yes
3.8.10.1	3.8.10.1	Verify correct breaker alignments to required AC, DC, and 120 VAC bus electrical power distribution subsystems.	7 days	yes	yes
Section 3.9, Refueling Operations					
3.9.1.1	3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours	yes	yes
3.9.2.1	3.9.2.1	Perform CHANNEL CHECK	12 hours	yes	yes
3.9.2.2	3.9.2.2	Perform CHANNEL CALIBRATION	18 months	yes	yes

ANO-1 SR #	NUREG-1430 R4 SR #	ANO-1 Surveillance Description {NUREG 1430 Description if no ANO-1 Surveillance}	ANO-1 Surveillance Frequency	SR Frequency Modified by TSTF-425	SR Frequency Modified in Proposed Amendment Request
3.9.3.1	3.9.3.1	Verify each required reactor building penetration is in the required status.	7 days	yes	yes
3.9.3.2	3.9.3.2	Verify each required reactor building isolation valve and each reactor building purge Isolation valve actuates to the isolation position	18 months	yes	yes
3.9.3.3	N/A	Perform CHANNEL CALIBRATION of reactor building purge exhaust radiation monitor	18 months	no	yes
3.9.4.1	3.9.4.1	Verify one DHR [Decay Heat Removal] loop is in operation	12 hours	yes	yes
3.9.5.1	3.9.5.1	Verify one DHR loop is in operation	12 hours	yes	yes
3.9.5.2	3.9.5.2	Verify correct breaker alignment and indicated power available to each required DHR pump	7 days	yes	yes
3.9.6.1	3.9.6.1	Verify refueling canal water level is \geq 23 feet above the top of irradiated fuel assemblies seated within the reactor pressure vessel	24 hours	yes	yes
Section 5.5, Programs and Manuals					
TS 5.5.2.b	N/A	The program shall include the following: Integrated leak test requirements for each system at least once per 18 months.	18 months	no	yes
TS 5.5.5.d	TSTF-425 SR 3.7.10.4	Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTR, at a Frequency of 18 months on a STAGGERED TEST BASIS.	18 months on a STAGGERED TEST BASIS	yes	yes
TS 5.5.13.c	N/A	Total particulate concentration of the fuel oil is \leq 10 mg/l when tested every 31 days based on ASTM D 2276, Method A.2 or A.3;	31 days	no	yes
5.5.8	5.5.20	N/A Surveillance Frequency Control Program	N/A	yes	yes