Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802
Tel 479-858-3110

Richard L. Anderson
ANO Site Vice President

10 CFR 50.90

2CAN021802

February 6, 2018

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 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 2
Docket No. 50-368
License No. NPF-6

REFERENCE: NUREG-1432, Standard Technical Specifications Combustion Engineering Plants, Revision 4, April 2012 (ML12102A165)

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 2 (ANO-2).

The proposed amendment would modify ANO-2 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 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 contains the 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-2 TS surveillance requirement numbers to the NUREG-1432 (Reference) TS surveillance requirement numbers.

Entergy requests approval of the proposed license amendment by February 2, 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 the designated Arkansas State Official.

If there are any questions or if additional information is needed, please contact Stephenie Pyle at 479-858-4704.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on February 6, 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. ANO-2 to NUREG 1432 SR Cross-Reference

cc: Mr. Kriss M. Kennedy
Regional Administrator
U. S. Nuclear Regulatory Commission
RGN-IV
1600 East Lamar Boulevard
Arlington, TX 76011-4511

NRC Senior Resident Inspector
Arkansas Nuclear One
P. O. Box 310
London, AR 72847

U. S. Nuclear Regulatory Commission
Attn: Mr. Thomas Wengert
MS O-08B1
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Mr. Bernard R. Bevill
Arkansas Department of Health
Radiation Control Section
4815 West Markham Street
Slot #30
Little Rock, AR 72205

ATTACHMENT 1 to

2CAN021802

DESCRIPTION AND ASSESSMENT

1.0 DESCRIPTION

The proposed amendment would modify 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 - Risk Informed Technical Specification Task Force (RITSTF) Initiative 5.” Additionally, the change would add a new program, the Surveillance Frequency Control Program, to TS Section 6.0, “Administrative Controls.”

The changes are consistent with NRC approved Industry/TSTF Standard Technical Specification (STS) change TSTF-425, Revision 3, (ML090850642). The Federal Register Notice published on July 6, 2009, 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 with regard to the Arkansas Nuclear One, Unit 2 (ANO-2) probabilistic risk assessment (PRA) technical adequacy consistent with the requirements of Regulatory Guide (RG) 1.200, Revision 1 (ML070240001), Section 4.2, and describes any PRA models without NRC-endorsed standards, including documentation of the quality characteristics of those models in accordance with RG 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-2 and justify this amendment to incorporate the changes to the ANO-2 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, including differing TS surveillance numbers.

1. The ANO-2 TS were based on the standard TS at the time of issue, which did not contain Bases as comprehensive as those in NUREG-1432. Therefore, many of the Bases mark-ups in TSTF-425 are not applicable to the ANO-2 TS. The proposed Bases changes in Attachment 4 revise only those Bases that currently discuss surveillance frequencies. This deviation is similar to that provided by the Florida Power and Light Company for the Turkey Point Nuclear Generating Units 3 and 4 (ML14105A042) as part of Amendments 263 and 258, respectively (ML15166A320). This is an administrative deviation from TSTF-425 with no impact on the NRC model SE dated July 6, 2009 (74 FR 32001).

2. To reflect the proposed change allowing instrument function frequencies to be controlled in accordance with the Surveillance Frequency Control Program (SFCP), ANO-2 TS Table 1.2, Frequency Notation, has been updated to include the notation “SFCP.” In addition, the following notations and associated frequencies have been deleted which are no longer used in the TSs: S, D, W, M, Q, TA, SA, and R. Note that ANO-2 TS Page 1-7 is currently blank and is being deleted. Subsequently, TS Page 1-8 (containing Table 1.1, Operational Modes) is renumbered as 1-7 and the revised Table 1.2 is moved to the same page as Table 1.1. In this respect, TS Pages 1-8 and 1-9 are deleted and removed from the TSs. TS Page 1-7 includes amendment numbers in the footer relevant to the three pages (Pages 1-7, 1-8, and 1-9) for historical tracking purposes. These are administrative deviations from TSTF-425 with no impact on the NRC model SE dated July 6, 2009 (74 FR 32001).

   In some instances, a table notation or footnote in the ANO-2 TSs specifies the frequency for an SR, where the frequency is being proposed for relocation to the SFCP. Entergy proposes to include the use of "SFCP" as a frequency notation in the tables and/or footnotes, where applicable, that are linked to affected instrumentation SRs (see ANO-2 TS Tables 4.3-1, 4.3-2, 4.3-3, 4.3-6, 4.3-10, and SR 4.4.12.2). This is an administrative deviation from TSTF-425 with no impact on the NRC model SE dated July 6, 2009 (74 FR 32001).

3. The approved programs for ANO-2 are described in Section 6.0, "Administrative Controls," of the ANO-2 TSs (the equivalent STS section is 5.0). Therefore, the SFCP requirements specified in TSTF-425 are proposed to be added to TS Section 6.0, TS 6.5.18. This is an administrative deviation from TSTF-425 with no impact on the NRC model SE dated July 6, 2009 (74 FR 32001).

4. Because the ANO-2 TSs are based on the standard TS at the time they were issued, the applicable surveillance requirements (SRs) and associated bases numbers differ from those presented in NUREG-1432 and TSTF-425 and are retained in this license amendment request. This is an administrative deviation from TSTF-425 with no impact on the NRC model SE dated July 6, 2009 (74 FR 32001).

5. For NUREG-1432 surveillances not contained in ANO-2 TSs, the corresponding mark-ups included in TSTF-425 for these surveillances are not applicable to ANO-2. This is an administrative deviation from TSTF-425 with no impact on the NRC model SE dated July 6, 2009 (74 FR 32001).

6. For ANO-2 plant-specific SRs that are not included in the NUREG-1432 markups provided in TSTF-425 (see Attachment 7 of this letter), Entergy has determined that the relocation of the frequencies for these plant-specific SRs is consistent with the intent of TSTF-425, Revision 3, and the NRC model SE dated July 6, 2009 (74 FR 32001). The scope exclusion criteria listed in the model SE are repeated below:

   - Frequencies that reference other approved programs for the specific interval (such as the Inservice Testing Program or the Primary Containment Leakage Rate Testing Program);

   - Frequencies that are purely event-driven (e.g., “Each time the control rod is withdrawn to the ‘full out’ position”);

- Frequencies that are event-driven but have a time component for performing the surveillance on a one-time basis once the event occurs (e.g., "within 24 hours after thermal power reaching ≥ 95% RTP");

- Frequencies that are related to specific conditions (e.g., battery degradation, age and capacity) or conditions for the performance of a surveillance requirement (e.g., "drywell to suppression chamber differential pressure decrease").

The subject plant-specific SR frequencies are periodic frequencies which do not meet the scope exclusion criteria. In accordance with TSTF-425, changes to the frequencies for these SRs would be controlled under the SFCP.

The SFCP provides the necessary administrative controls to require that SRs related to testing, calibration, and inspection are conducted at a frequency that ensures that the necessary quality of systems and components is maintained, that facility operation is maintained within safety limits, and that the limiting conditions for operation are met. Changes to frequencies in the SFCP are evaluated using the NRC-approved methodology and PRA guidelines contained in NEI 04-10, Revision 1. This deviation from TSTF-425 is consistent with other industry applications adopting TSTF-425 (e.g., Waterford Unit 3 License Amendment 249 – ML16159A419, Perry Unit 1 License Amendment 171 – ML15307A349, and Brunswick Units 1 and 2 License Amendments 276 and 304, respectively – ML17096A129).

7. Periodic frequencies associated with ANO-2 TSs 6.5.2 and 6.5.13 are included in the scope of this amendment that are not identified for relocation in TSTF-425, Revision 3.

   The first sentence of TS 6.5.2.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 6.5.13.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 is 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 6.5 frequencies are periodic frequencies and do not meet the scope exclusion criteria identified in Section 1.0, "Introduction," of the model SE. This change is similar to the program change described in Item 8 below.

   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 is within safety limits, and that the

limiting conditions for operation are 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.

8. TSTF-425 includes a relocation of the frequency for NUREG-1432, SR 3.7.11.4, associated with verifying the Control Room Emergency Air Cleanup System (CREACS) maintains 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 inleakage testing in accordance with the Control Room Habitability Program. The requirement to perform the relative pressure surveillance was included in the new NUREG 1432, TS 5.5.18, "Control Room Envelope (CRE) Habitability Program," as TS 5.5.18.d. ANO-2 adopted TSTF-448 in Amendment 288 dated October 2009 (ML082520574), designating the Control Room Habitability Program as TS 6.5.12 with the specific SR as TS 6.5.12.d. Therefore, the TSTF-425 SR 3.7.11.4 frequency change is being adopted in ANO-2 TS 6.5.12.d. This is an administrative deviation from TSTF-425 with no impact on the NRC model SE dated July 6, 2009 (74 FR 32001). In addition, on July 26, 2016, the NRC approved a similar SR frequency relocation (i.e., TS 6.5.17.d frequency) in Waterford Unit 3 TSTF-425 License Amendment 249 (ML16159A419).

9. The response time tests of the reactor protective instrumentation (i.e., 4.3.1.1.3) and engineered safety features actuation system instrumentation (i.e., 4.3.2.1.3) contain text describing what constitutes “N” for response time testing (similar to the TS Definition of Staggered Test Basis). This plant specific text identified for deletion in Attachment 3 is not specified the NUREG-1432 mark-ups provided in TSTF-425. However, text that describes what constitutes “N” is identified as Notes in other STS NUREGs (e.g., Note 2, NUREG-1434, SR 3.3.1.1.15, RPS Response Time test) and is identified for deletion in TSTF-425. For these plant-specific changes that are not contained in the NUREG-1432 mark-ups provided in TSTF-425, Entergy has determined that the changes are consistent with the intent of TSTF-425, Revision 3, and with the NRC model SE dated July 6, 2009 (74 FR 32001). The subject plant-specific SR frequencies do not meet the scope exclusion criteria identified in Section 1.0, "Introduction," of the model SE. Further, Section 1.0 of NEI 04-10, Revision 1, states, in part:

   NEI 04-10 Revision 1 contains new information…to address how Surveillances which are performed on a Staggered Test Basis are modeled in the risk assessment performed to support a change to the Frequency. This will allow licensees to add or remove the requirement to perform Surveillances on a Staggered Test Basis under the Surveillance Frequency Control Program.

   In accordance with TSTF-425, changes to the frequencies for these SRs would be controlled under the SFCP. The SFCP provides the necessary administrative controls to require that SRs 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 are evaluated using the NRC approved methodology and PRA guidelines contained in NEI 04-10, Revision 1. This deviation from TSTF-425 is similar to that of Millstone Unit 2 TSTF-425 application (ML15280A242 and ML14301A112).

10. Editorial changes are included in ANO-2 SRs 4.5.1.a.2 and 4.5.2.a by correcting valve numbering and clarifying that the requirement to verify power is removed (SR 4.5.2.a) applies only to the two motor operated valves (the third valve in the list is a manual valve). For example, 2CV-5101 (as currently listed) should be more accurately identified as 2CV-5101-1 (the -1 indicating vital red-train power supply). Table 5.2-2 of the ANO-2 SAR correctly illustrates these valve numbers. This is an administrative deviation from TSTF-425 with no impact on the NRC model SE dated July 6, 2009 (74 FR 32001).

11. Editorial changes are included in ANO-2 Specification 6.5.17, “Metamic Coupon Sampling Program.” Currently, this program contains reference to SR 3.0.2 and SR 3.0.3 which is NUREG 1432 numbering. These are corrected to refer to the equivalent ANO-2 TS numbering of SR 4.0.2 and SR 4.0.3. This is an administrative deviation from TSTF-425 with no impact on the NRC model SE dated July 6, 2009 (74 FR 32001).

Attachment 7 provides a cross-reference between the ANO-2 SRs frequencies included in this amendment request versus the NUREG-1432 SRs frequencies included in TSTF-425. This cross-reference is provided for information purposes only and is not intended to be a verbatim description of the TS SRs. Attachment 7 includes a summary description of the referenced ANO-2 TS SRs or the TSTF-425 (NUREG-1432) SRs, if there is no corresponding/similar ANO-2 SR. Some links to NUREG 1432 SRs are not a direct correlation, but are similar requirements (e.g., “verify dose equivalent XE-133” versus “verify gross specific activity” on a 7-day frequency (ANO-2 SR 4.4.8.1 and NUREG 1432 SR 3.4.16.1, respectively)).

The Attachment 7 cross-reference highlights the following:

a. ANO-2 surveillances with plant-specific SR numbers and corresponding or similar NUREG-1432 SRs included in TSTF-425

b. NUREG-1432 SRs included in TSTF-425 that are not contained in the ANO-2 TSs

c. ANO-2 plant-specific surveillances that are not contained in NUREG-1432 and, therefore, are not included in the markups in TSTF-425

d. Prior NRC approvals for similar SR changes not included in TSTF-425:

    i. Milestone 2 – ADAMS Accession No. ML15280A242

    ii. St Lucie 1 and 2 – ADAMS Accession No. ML15127A066

    iii. Waterford 3 – ADAMS Accession No. ML16159A419

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

3.0 REGULATORY ANALYSIS

3.1 Applicable Regulatory Requirements/Criteria

A description of the proposed changes and their relationship to applicable regulatory requirements is provided in TSTF-425, Revision 3, and the NRC model safety evaluation (SE) 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-2.

3.2 No Significant Hazards Consideration

Entergy Operations, Inc. (Entergy) has reviewed the proposed no significant hazards consideration determination (NSHC) published in the 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 2 (ANO-2) 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 Millstone 2 on October 29, 2015 (ML15280A242), St Lucie 1 and 2 on June 22, 2015 (ML15127A066), and Waterford 3 on July 26, 2016 (ML16159A419).

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 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 ANO-2, and the determination is hereby incorporated by reference for this application.

ATTACHMENT 2 to

2CAN021802

DOCUMENTATION OF PRA TECHNICAL ADEQUACY

DOCUMENTATION OF PRA TECHNICAL ADEQUACY

TABLE OF CONTENTS

1. PURPOSE 2

2. SCOPE 2

2.1 Surveillance Frequency Change Process 3

2.2 Technical Adequacy of a PRA 4

3. ANO-2 PRA TECHNICAL ADEQUACY 5

3.1 Discussion 5

3.2 ANO-2 Internal Events and Internal Flooding PRA Model 6

3.2.1. Plant Changes Not Yet Incorporated 6

3.2.2. Peer Review Facts and Observations (F&Os) 6

3.2.3. Consistency with Applicable PRA Standards 7

3.3 ANO-2 Fire PRA Model 40

3.3.1. Plant Changes Not Yet Incorporated 40

3.3.2. Peer Review Facts and Observations 40

3.3.3. Consistency with Applicable PRA Standards 41

3.4 Identification of Key Assumptions 80

3.5 External Events and Shutdown Considerations 80

4. CONCLUSIONS 81

5. REFERENCES 81

LIST OF TABLES

Table 1 List of Finding F&Os Against the ANO-2 Internal Events and Internal Flooding Models 8

Table 2 List of Finding F&Os against the ANO-2 Fire PRA Model 43

Table 3 List of SRs Assessed as CC-1 in the ANO-2 Fire PRA Model 78

1. PURPOSE

The purpose of this report is to document the technical adequacy of the Arkansas Nuclear One, Unit 2 (ANO-2) 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-2 intends to follow the guidance provided in NEI 04-10, Revision 1 (Reference 2), in evaluating proposed surveillance test interval (STI) changes (also referred 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 14), 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-2 PRA model Revision 5p00, 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-2 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-2 fire PRA model (self-approval model) update was completed in 2017. This model was constructed to meet the requirements of NUREG/CR-6850 (References 15 and 16). 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, as well as site specific 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 reassures 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-2 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 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 thru 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-2 PRA TECHNICAL ADEQUACY

3.1 Discussion

The ANO-2 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-2 PRA model 5p00 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-2 internal events and internal flooding models.

The ANO-2 fire PRA model (self-approval model) update based on the ANO-2 full power internal events (FPIE) 5p00 model was completed in 2017 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-2 fire PRA model.

3.2 ANO-2 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 Capability 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 plant and procedure review was performed as part of the upcoming ANO-2 PRA model revision 6 (Reference 8). Several engineering changes (ECs) were identified as being PRA significant, and MCRs had been initiated to track the potential changes to the PRA model. ECs and corresponding MCR numbers are reported in Table 6.1 and Attachment 1 of Reference 8, as well as the potential impact on the PRA model.

3.2.2 Peer Review Facts and Observations (F&Os)

The ANO-2 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-2 model:

- An industry peer review of the ANO-2 PSA model Revision 3p0 was conducted by the Combustion Engineering Owners Group (CEOG) in 2002 (Reference 5). The peer review concluded that there were several areas where the ANO-2 model needed improvement. The ANO-2 PSA model updates Rev 3p01, 3p02, 4p00, and 4p01 (4p01 completed in October 2006) addressed the Finding-level F&Os from this peer review.
- In preparation for ANO-2's transition to National Fire Protection Association (NFPA) 805 standard, a gap assessment of the ANO-2 probabilistic safety assessment (PSA) 4p01 internal events PRA model was completed. The gaps impacting the fire PRA were closed to meet the NFPA transition schedule. The ANO-2 Internal Events PSA model was updated (Rev 4p02 completed December 2008) to meet the RG 1.200, Revision 1, standards.
- In July 2008, a peer review of the Rev 4p02 ANO-2 PSA model was performed and documented in LTR-RAM-II-020 (Reference 6). This peer review documented fifty-nine (59) new F&Os including thirty-three (33) Findings and twenty-six (26) Suggestions. Most of the findings pertained to documentation issues. The conclusion of the review was that the ANO-2 PRA substantially met the ASME PRA standard at Capability Category II or better.
- In September 2017, a self-assessment of all ANO-2 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-2 internal events model revision 5p00 was approved in 2016. This is the current PRA model as stated in Section2. The peer review findings and the associated resolutions, 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 1.

The current ANO-2 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-002 (ENERCON report) (Reference 7). The results of the assessment show that approximately 90% of the internal flooding Supporting Requirements (SRs) were met at the Capability Category II level. Twenty-three (23) F&Os were issued during this peer review, including eleven (11) 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 1. 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-2 PRA model revision 5p00 meets the ASME/ANS PRA standard (Reference 4) Capability Category II of the SRs. 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 A2-5585 (missing table from Success Criteria, Accident Sequence and System Analysis packages). This documentation issue is expected to be resolved as part of the model update.

The latest full-scope peer review for ANO-2 was conducted in July 2008 using RG 1.200, Revision 1. Since then, model revision 5p00 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-2 MCRs related to the peer review F&Os was performed. Finding-level F&O MCRs related to the internal event and internal flooding model are listed in Table 1, along with the respective disposition/resolution, if resolved, and the impact on the applications.

Table 1

List of Finding F&Os against the ANO-2 Internal Events and Internal Flooding Models

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A2-3093	Resolved	IE-C3	RG1.200 Peer Review F&O IE-C3-01, Finding Issue: ANO-2 explicitly calculated the total reactor critical years as total reactor critical hours divided by 8766 hours per year and used this to calculate the Initiating Events Frequencies (IEFs). However, there is no evidence that ANO-2 adjusted these IEFs to reflect average plant availability. This is, in essence, equivalent to assuming that the plant operates at full power all year. ANO-2 needs to adjust the initiating event frequencies to account for average plant availability. Adjust the initiating event frequencies to account for the average plant availability.	Data sheets were set up to automatically adjust for capacity factor. Documented in Initiating Events (IE) notebook.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3094	Resolved	IE-C10	RG1.200 Peer Review F&O IE-C10-01, Finding No comparison of results to generic data sources was provided with a discussion and explanation of the differences. This is a requirement to meet IE C-10 and is important to assessing the validity of the initiating event frequency results. The ISLOCA IEF needs to be reviewed, compared and understood. This IEF value is very low. Compare the results to the generic values in NUREG/CR-5750 and NUREG/CR-6928, and provide and explain any significant differences.	Table 13 of PSA-ANO2-01-IE compares the IE frequencies to generic values.	This documentation 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	Importance to Application
A2-3095	Resolved	IE-C12 LE-D3	RG1.200 Peer Review F&O IE-C12-01, Also affects SR LE-D3, Finding Some of the components in the ISLOCA fault tree model appear to have incorrect mission times. The Low Pressure Safety Injection (LPSI) Motor Operated Valve (MOV), e.g. 2CV-5017 rupture, has a mission time of 36 hours. However, the mission time should probably be 8760 hours because the MOV rupture is not likely to be annunciated in the control room as assumed. Therefore, it could potentially be in an undetected failed state for an extended period. The same comment may apply to the second check valve. It could be potentially in an undetected failed state for an extended period. Reconsider and change the mission times for the time the downstream valves can be in an undetected state.	In addition to the comment in this MCR, the peer review team noted that the failure rate should be based on Large Internal Leakage rather than Large External Leakage. Therefore, the probability of Type Code MVG V 2 is changed from 1.0E-09/hour to 3.0E-09/hour and the probability of Type Code CVG V 2 is changed from 2.0E-09/hour to 3.0E-08/hour. Based on a review of the ISLOCA events, the mission times for several events were changed from 36 hours to 18 months. In addition, the initiator events were changed to EQU gates with new events tied to the Type Code file.	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	Importance to Application
A2-3096	Resolved	IE-D1	RG1.200 Peer Review F&O IE-D1-01, Finding Some of the documentation is not adequate to meet this requirement. The following items should be addressed: (1) What is the basis for the 80/20 split between reactor trip and turbine trip events? (Assumption 8, Section 2.2 and Section 5.1 page 19). (2) Where is the documentation for the small and medium break sizes that are used in the model? The lower limit for the large LOCA break size has a reference (see Table 2). (3) Appendix C contains calculations for loss of feedwater/condensate (T2). Page 48 contains the Bayesian update for this event. Recommend explaining how these are used in the PRA model and which is used in the base model. The value in Table 7 appears to be from the Bayesian update which is inconsistent with the discussion in Section 5.3.1. (4) Table 7: It's not clear where the frequencies for T500KV and TST3 come from. This should be explained. (5) The RR file with the quantification information contains frequencies for the following events that are not in the IE documentation: T3SD, T3SW, TSDCA, TSDCAML, TSDCB, TSDCBML, TSDCISO, V5+, and VS+.	(1) The updated data split reactor trip and turbine trip events using RADS database. (2) The success criteria notebook provides documentation of the small and medium LOCA equivalent break size (3) The value based on IEFT Appendix C is different from the number based on actuarial calculation. Compensator BE to adjust the IEFT estimation is used to equalize the IEFT result with the actuarial calculation. Provided discussion on this. (4) %T500 is documented in LOSP notebook draft ECH-NE-12-00099. The update for %T500 includes data up to 2011. %TST3 (Startup Transformer 3 fails as an Initiating Event) is documented in the updated IE notebook draft.	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	Importance to Application
A2-3096 (cont.)	Resolved	IE-D1	(6) Loss of Lake Dardanelle IE - need to document the change from 2E-04/yr to 1E-05/yr - it should be explained how the IE frequency for this event was reduced to 1E-05/yr. Provide documentation for the above six items.	(5) T3SD, T3SW, TSDCA, TSDCAML, TSDCB, TSDCBML, TSDCISO are not in the mitigating model so this is out of scope. V5+ and VS+ are documented in the updated IE Notebook. All events except T3SW are part of the Shutdown PSA for ANO-2. T3SW is documented in PSA-ANO2-01-IE-01 (6) See MCR A2-4697.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3099	Resolved	AS-A4	RG1.200 Peer Review F&O AS-A4-01, Finding Even though some operator actions required to achieve the identified success criteria are mentioned in portions of the initiating event analyses, these operator actions are not consistently identified and documented. Entergy should explicitly identify all operator actions needed to achieve the success criteria for each of the key safety functions defined for the modeled initiating events.	Operator actions were included in the Success Criteria (SC) and Accident Sequence (AS) notebooks.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3100	Resolved	AS-A5	RG1.200 Peer Review F&O AS-A5-01, Finding There is no reference to the System design, Emergency Operating Procedures (EOPs), or Abnormal Operating Procedures (AOPs) in the accident sequence notebook. It would be helpful if the EOP or abnormal procedure used for each accident sequence was noted. Add a table showing the EOPs or abnormal procedures used for each accident sequence.	EOPs and AOPs are documented in related tables in the SC notebook	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	Importance to Application
A2-3101	Resolved	AS-A10	RG1.200 Peer Review F&O AS-A10-01, Finding The operation actions are not specified in either the accident sequence detailed description or the event tree. An example would be a detailed discussion of the once through cooling and the operator actions required. Specify the operator actions in either the accident sequence detailed description.	Operator actions were included in the SC and AS notebooks. Detailed discussions regarding operator actions are found in the HRA notebook.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3102	Resolved	AS-B1	RG1.200 Peer Review F&O AS-B1-01, Finding The special initiators do not address the impact of these initiators on the mitigating systems. Proposed resolution as depicted in ILRT extension RAI to NRC: ANO-2 uses a linked fault tree approach where the initiating events are placed in the tree with the appropriate system model. A table will be added to the accident sequence report to show how the IE fails both the front line and support systems. This finding will not impact any applications.	A table was added to the AS notebook to address the issue.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3103	Resolved	AS-B2	RG1.200 Peer Review F&O AS-B2-01, Finding The dependencies are not addressed in the Accident Sequence notebook. This is especially true of operator actions and how the failure of an operator action would affect subsequent operator actions.	The Event Trees and discussion address required systems. Operator actions and dependencies are included in the HRA notebook. System dependencies are documented in the systems notebooks.	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	Importance to Application
A2-3104	Resolved	AS-B3	RG1.200 Peer Review F&O AS-B3-01, Finding No assumption or statement is made that plant equipment will perform in the environment for which it was designed. There also was no evidence that equipment not specified in the Safety Analysis Report (SAR) for accident mitigation but still credited in the PRA were reviewed for environmental affects. This could be an assumption that the equipment meets the environmental qualification. Equipment that is not environmentally qualified need to be analyzed on the impact they have on the applicable accident sequence.	A section was added to the accident sequence analysis to document environmental considerations.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3105	Resolved	AS-B6	RG1.200 Peer Review F&O AS-B6-01, Finding This SR was not met because there was no discussion of the following: changes in environmental conditions, shifting of the Condensate Storage Tanks (CSTs), and operator actions. Questions were raised in that ANO-2 uses 40 minutes as the time that the Reactor Coolant Pump (RCP) can run without Component Cooling Water (CCW) cooling. The industry practice (WCAP-16175) uses 20 minutes. This difference should be analyzed and resolved. Discuss the following: changes in environmental conditions, shifting of the CSTs, and operator actions.	20 minutes instead of 40 minutes has been used for RCP running without CCW cooling in AS and SC notebook. 20 Minutes has been used in the PRA, updated in the HRA spreadsheets and documented in the AS and SC notebooks. Environmental issue was discussed in system documentation.	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	Importance to Application
A2-3107	Resolved	AS-C2	RG1.200 Peer Review F&O AS-C2-01, Finding The documentation does not show that the items in this SR have been addressed. Address all the items in this SR in the assumption section.	All the items in the SR were incorporated in the Accident Sequence documentation.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3116	Resolved	SY-A4	RG1.200 Peer Review F&O SY-A4-01, Finding Walkdowns will need to be performed in order to support National Fire Protection Association (NFPA) 805 Fire PRA and Flooding initiator. This SR can be accomplished during this process.	Fire PRA walkdowns for NFPA-805 are documented in ANO Calculation CALC-08-E-0016-01, Appendix D. While this walkdown documents some spatial and environmental impacts due to a fire in the area, the documentation for internal events is somewhat different. MCR A2-3119 addresses the documentation of walkdowns and operator interviews for internal events.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3118	Resolved	SY-A8	RG1.200 Peer Review F&O SY-A8-01, Finding The EDG air start system is included in the component boundary of the EDG for failure rate and common cause but is still modeled in the fault tree with non-zero probabilities.	The Emergency Diesel Generator (EDG) air start system is considered part of the EDG component boundary as discussed in NUREG/CR-6928 and Echelon Calculation PRA-ES-01-003. Therefore, the diesel air start basic events will be set to a probability of 0.0 in the ANO-2 database. The events will be retained in the model for EOOS purposes.	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	Importance to Application
A2-3119	Resolved	SY-B8	RG1.200 Peer Review F&O SY-B8-01, Finding No documentation of spatial and environmental hazards assessment was found. LKB Comments 9/4/2014 - MCRs A2-2462 and A2-2466 is Resolved to this MCR due to similar PSA Gap Analysis finding. Ensure that any additional issues from MCR A2-2462 and A2-2466 are addressed as part of this MCR.	System walkdowns and interviews were performed during the ANO-2 PRA Model 5p00 update and documented in the system notebooks. All PRA systems that are normally accessible were physically walked down to assess both special and environmental hazards. MCR A2-5337 documents the plan to walkdown systems located inside containment. For each walked down system, an interview was also conducted with the responsible system engineer to discuss the state of each system.	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	Importance to Application
A2-3121	Unresolved	SY-B15	RG1.200 Peer Review F&O SY-B15-01, Finding There was no indication in any of the three systems reviewed that an assessment was completed to determine if the SSCs will be required to operate in conditions beyond their environmental qualifications.	A discussion of how the system will respond in harsh environments has been added in the Rev 6 model documentation. In order to address the generic review/comment, this discussion was expanded to state the following: The internal flooding analysis performed walkdowns of the steam and water lines in the plant, and identified the PRA credited equipment in each room that was potentially susceptible to flooding, spray, and a steam environment due to a pipe break in the room. The internal flood analysis includes scenarios that model these impacts specifically, so they are not duplicated in the internal events PRA. Refer to the ANO-2 internal flood analysis for additional details.	This is a documentation issue. Additional information under “Harsh Environments”, as applicable, was added to the system notebooks being revised for the ongoing model revision 6 update.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A2-3121 (cont.)	Unresolved	SY-B15	RG1.200 Peer Review F&O SY-B15-01, Finding There was no indication in any of the three systems reviewed that an assessment was completed to determine if the SSCs will be required to operate in conditions beyond their environmental qualifications.	The ability of the system equipment to continue to operate following a loss of heating, ventilation, and air condition (HVAC) in the areas where system equipment is installed is addressed by either by modeling a loss of HVAC as a failure of the impacted equipment, or justifying the basis for why the equipment is reasonably expected to continue to operate with HVAC. The original bases for why a loss of HVAC does not fail the equipment in the system is documented in Section #.2.2, if applicable. The SC notebook may revise these assumptions if additional room heat-up analyses are performed which provide additional insights into the impact of a loss of HVAC on the system equipment	This is a documentation issue. Additional information under “Harsh Environments”, as applicable, was added to the system notebooks being revised for the ongoing Revision 6 model update.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A2-3124	Resolved	HR-C2	RG1.200 Peer Review F&O HR-C2-01, Finding CC-I was assessed to be met, but there is no direct evidence that ANO-2 evaluated plant-specific or generic operating experience to check for other pre-initiators. Documentation of a review of plant specific information or industry Licensee Event Reports (LERs) from other similar plants and incorporating this information into the HRA assessment is required to receive a CC-II/III rating. Incorporate an assessment of the plant-specific or generic operating experience information into the HRA assessment.	A review of ANO-2 Condition Reports (CRs) was performed. The resulting search yielded 39 CRs. The review of these CRs showed most of the events involved instruments that were slightly out of calibration. Other events resulted in equipment that remained operable and thus would not be PSA applicable. The CR review spreadsheet is included in App I of PSA-ANO2-01-HR	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	Importance to Application
A2-3125	Resolved	HR-D3	RG1.200 Peer Review F&O HR-D3-01, Finding Not assessed consistent with CC II since the evaluation does not provide an assessment of the quality of the procedures or the quality of the human-machine interface. Provide an assessment of the quality of the procedures and the human-machine interaction. If this has been done, provide the documentation.	A review of quality of ANO-2 procedures was performed. A review of CRs identified two CRs associated with procedure quality. CR-ANO-C-2009-00716 and CR-ANO-C-2010-01876 were written to correct issues with procedure quality. CR-ANO-C-2009-00716 is based on an INPO Area for Improvement and includes more than 200 corrective actions that have been completed. CR-ANO-C-2010-01876 provides 4 recommendations from an INPO Assist Visit that resulted in an additional 31 corrective actions. Text was added to the HRA Notebook regarding general procedure quality. A discussion was also added to the section on Quantification Using the Cause-Based Approach to highlight that it does consider specific aspects of the quality of the procedures (in decision trees e-g) and human-machine interface (in decision trees a-d).	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	Importance to Application
A2-3126	Resolved	HR-D6	RG1.200 Peer Review F&O HR-D6-01, Finding ANO2 uses the HRA Toolbox for quantifying their pre-initiator HEPs. For the pre-initiator HEPs, ANO2 basically uses the ASEP approach and treats the ASEP Basic HEPs as means with the associated error factors. However, as defined on page xv of NUREG/CR-4772, the ASEP BHEP values are medians for a log-normal distribution. Thus, the treatment of the BHEP values for the pre-initiators is mathematically incorrect.	A conversion formula was added to the bottom of each of the individual hfe_a events and to the a_template templates in the HRA Toolbox so that future pre-accident HRAs will have the conversion from medians to means. The median to mean conversion is shown in cell I64 and the mean probability is shown in cell F64. In addition, cell AH41 was changed to equal F64 to provide the mean value to the hfe_summary.xls. The values are changed in PRA-A2-01-003S03, Revision 2.	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	Importance to Application
A2-3127	Resolved	HR-G6	RG1.200 Peer Review F&O HR-G6-01, Finding This SR requires a check of consistency of the post-initiator HEP quantification. This requires a review of the HFEs and their final HEPs relative to each other to check their reasonableness. There is no evidence that this consistency check has been done. If done this should be documented, if not done this should be completed. This can be addressed by adding an explicit process for reviewing the HEPs for internal consistency with respect to scenario, context, procedures and timing. Specifically this can evaluate the HEPs with respect to certain expected patterns such as increasing HEPs with decreasing time available, increasing HEPs with increasing stress levels, and increasing HEPs with increasing complexity of the procedures for accomplishing the desired successful outcome. A statement that such an evaluation was performed and, where there were deviations from the expected patterns and either provides a basis for the deviation or what was done to correct it.	HRA events were reviewed for consistency and were documented in the HRA notebook.	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	Importance to Application
A2-3128	Resolved	HR-G9	RG1.200 Peer Review F&O HR-G9-01, Finding This requires the use of means values. NUREG-1278 contains median values that do not appear to be converted to means before being used in the ANO2 PRA. For example, spread sheet used for HRA at ANO2.	The execution errors in hfe_cp.xls are shown in cells CC16:CC61. These values are medians. To convert to means, the median values were moved to cells CE16:CE61. The error factors from NUREG/CR-1278 were added in cells CF16:CF61. Cells CC16:CC61 are changed to =CE??*EXP(POWER(1/1.645*LN (K61),2)/2) where "??" is the associated median value in the accompanying CE cell. This change was made in all hfe_cp worksheets including cp_template, in the HRA Toolbox, which will ensure that future procedural operator actions will be calculated based on means rather than medians. The values are changed in PRA-A2-01-003S03, Revision 2.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3129	Resolved	DA-A1a	RG1.200 Peer Review F&O DA-A1a-01, Finding Boundary developed for EDG starting air was outlined in PRA-ES-01-003 included the start air system inside the component boundary. The CAFTA model had the starting air modeled with Basic Events (BEs) set greater than zero, effectively placing the starting air outside the component boundary. See F&O SY-A8-01 for details.	Starting air basic events were set to 0.0 in basic event file but left in model to allow EOOS modeling if needed.	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	Importance to Application
A2-3130	Resolved	DA-C10	RG1.200 Peer Review F&O DA-C10-01, Finding CAT I given based on information listed in Procedure PRA-A2-01-003S05 does not address decomposing the component failure mode into sub-elements (or causes) that are fully tested, then using tests that exercise specific sub-elements in their evaluation. May be over-counting demands and run-hours for component boundaries that are not tested during evolution. Need to review component boundaries and tests counted in data collection to ensure that one sub-element does not have many more successes than another. Update procedure CE-P-05.07 with process details that ensure the requirements described in CAT II/III are met.	Documented in the Data Analysis Notebook. Data provided addresses the component boundaries and re-testing.	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	Importance to Application
A2-3131	Resolved	DA-C12	RG1.200 Peer Review F&O DA-C12-01, Finding Procedure PRA-A2-01-003S05 addresses evaluating maintenance outage as a function of plant status. CAT I given since there is no evidence of INTERVIEW the plant maintenance and operations staff to generate estimates of ranges in the unavailable time per maintenance act for components, trains, or systems for which the unavailabilities are significant basis events. As a suggestion, need to include interviews and shared equipment between ANO-1 and ANO-2 (i.e., air compressors) in procedure PRA-A2-01-003S05. Need to document interviews in order to meet Category II/III. Update procedure CE-P-05.07 with process to perform interviews with plant maintenance and operations staff to generate estimates of ranges in the unavailable time per maintenance act for components, trains, or systems for which the unavailabilities are significant basis events. Document interviews.	The finding noted pertained to a prior version of the ANO-2 data analysis dated May 2008. The ANO-2 data analysis was most recently updated in May 2015, and is undergoing current update, where actual plant data were obtained for most risk significant unavailability terms in the model. The other risk significant terms used estimates that were provided by the PRA staff and the Maintenance Rule Coordinator. See Section 1.4 and Table 7 of PSA-ANO2-01-DA, Revision 1. Procedure CE-P-05.07 has been superseded by fleet procedure EN-DC-151. In addition, engineering guide EN-NE-G-007 includes guidance to obtain and document data from plant personnel.	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	Importance to Application
A2-3133	Resolved	IF-A1	RG1.200 Peer Review F&O IF-A1-01, Finding, Applicable to all IF SRs. At the time of the peer review, the ANO-2 IF analyses had not been completed to the point that it could be reviewed. Entergy intends to use the same IF methodology for all three of their PWRs with the Waterford-3 plant being the lead plant. The Waterford-3 IF analysis had been completed. Entergy requested that the peer review team review the IF methodology for Waterford to confirm that the methodology met the standard. Entergy needs to complete the ANO-2 IF analyses using the Waterford-3 methodology. Entergy will need to specifically address dual unit issues for ANO-1 and ANO-2.	An internal flood analysis (IFA) update was completed and met standard requirements. The analysis incorporates updated walkdowns and quantifies scenarios that were screened out in previous revisions.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3134	Resolved	IF-C2c	RG1.200 Peer Review F&O IF-C2c-01, Finding Equipment height off floor appears to be not recorded for most of the equipment on the walk down sheets. For example, flood area TB-15-250 walkdown sheet on page 460 only 3 of 26 items listed include a height. Spatial location in the area (for example height off floor) and any flooding mitigative features (e.g., shielding, flood or spray capability ratings) is not recorded for most of the PRA components listed on the walkdown sheets. Therefore, for a particular flood height in a room, it is not clear whether or not a component is affected. Complete the walkdown sheets.	Flooding documentation forms have input for height above floor, but only filled them in for cases where the components might be susceptible to inundation from flood sources. Therefore, no additional information needed.	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	Importance to Application
A2-3135	Resolved	IF-C3	RG1.200 Peer Review F&O IF-C3-01, Finding The walk down sheets identify the components located inside the flood area. This SR requires that components in a flood area be identified and include whether the component is susceptible to failure by submergence or spray. The walkdown sheets are formatted to allow recording whether or not the component is vulnerable to spray. Only several walkdown sheets have the column filled out for vulnerability to spray. It is not clear whether blanks indicate not susceptible to spray or not.	Flooding documentation forms have input for height above floor, but only filled them in for cases where the components might be susceptible to inundation from flood sources. Therefore, no additional information needed.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3137	Unresolved	IF-D6	RG1.200 Peer Review F&O IF-D6-01, Finding Operator error contributions to flooding are discussed at a very high level. However, basically the only floods considered were catastrophic failures. The flood scenario frequencies were quantified using generic pipe rupture data and plant-specific pipe length. The generic flood frequency sources do not include floods caused by human actions during maintenance. While the operator induced floods may be less severe than the catastrophic pipe failure floods, the frequencies will be higher so should be considered explicitly.	The IFA was upgraded in 2016 including the treatment of human-induced flooding	Human-induced flooding represents a small portion of the internal flooding events. A very small change to flooding results is expected. Impact of this finding is expected to be assessed in a case-by-case STI evaluations.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A2-3142	Resolved	QU-F4	RG1.200 Peer Review F&O QU-F4-01, Finding Selection process for determining important assumptions and sources of uncertainty was not delineated. LKB Comment 9/8/2014 - MCR A2-2464 is closed to this MCR. Ensure that the issues from MCR A2-2464 are addressed when closing this MCR.	Appendix 1 of the Sensitivity and Uncertainty includes a review of the sources of uncertainty using EPRI 1016737.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3144	Unresolved	LE-D1b	RG1.200 Peer Review F&O LE-D1b-01, Finding There is no evidence of an evaluation of the impact of the accident progression conditions on containment seals, penetrations, etc. The model this is based on is related to NUREG/CR-6595 so consistency with NUREG/CR-6595 meets CC-I, but there is no discussion of the accident progression conditions on these elements. Provide a discussion or assessment of the accident progression conditions on the containment conditions noted in the SR.	The LERF model and documentation is being updated as part of the Rev 6 update to include impact of accident progression conditions on containment seals, penetrations, etc.	Minimal or no impact on the results since assumptions are conservative. This Unresolved F&O does not impact the PRA quality. However, impact of this finding is expected to be assessed in case-by-case STI evaluations, for surveillances related to containment boundaries.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A2-3145	Unresolved	LE-D6	RG1.200 Peer Review F&O LE-D6-01, Finding Containment isolation is addressed by top event (question) 3. This is based on a calc that is noted not to have been maintained up to date. Since it has not been maintained up-to-date, there is no confidence that the analysis represents a realistic assessment; therefore, this does not meet CC II. The containment isolation calc needs to be updated or demonstrated (confirmed) to be up to date. This should include an assessment of the containment penetrations to provide an assessment of the total number of penetrations required to provide a realistic evaluation of containment isolation reliability.	Containment isolation model is currently being updated as part of the Rev 6 model update	Minimal or no impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluations, for surveillances related to containment boundaries. The LERF documentation and system notebooks and model are being updated as part of the Rev 6 model revision.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A2-3147	Resolved	LE-E4	RG1.200 Peer Review F&O LE-E4-01, Finding Although the majority of the SR requirements in these three top high level requirements are met, there is no indication that dependencies between multiple HFEs have been addressed. The Level 1 assessment completed an evaluation of the dependencies between human actions in the model. A similar analysis should be completed for the human actions in the Level 2 analyses and between the Level 2 and Level 1 analyses to ensure all HEP dependencies are identified and addressed appropriately.	Fixed recovery file for LERF to deal with potentially dependent HEP combinations. About 5 new HEP dependency combinations were identified and entered in the HRA tool box using the method of iterative review of the top 100 cutsets. Assumption 35 was added to the LERF notebook that states the following: "Assumption: HRA dependencies among Level 1 and Level 2 human failure events and among multiple Level 2 human failure events does not require additional modeling. Justification: The Level-1 and Level-2 operator actions were assumed to be independent, because they are performed by different plant personnel and at different plant conditions and are separated by time. Dependencies between Level-2 operator actions were also assumed to be independent of each other.	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	Importance to Application
A2-3147 (cont.)	Resolved	LE-E4	RG1.200 Peer Review F&O LE-E4-01, Finding Although the majority of the SR requirements in these three top high level requirements are met, there is no indication that dependencies between multiple HFEs have been addressed. The Level 1 assessment completed an evaluation of the dependencies between human actions in the model. A similar analysis should be completed for the human actions in the Level 2 analyses and between the Level 2 and Level 1 analyses to ensure all HEP dependencies are identified and addressed appropriately.	Most of the Level 2 operator actions are included in the split fraction for various Level 2 phenomena; these phenomenological events are conservatively modeled based on the uncertainty of post-core damage activities. Those that are explicitly modeled were judged to be independent, because they address unique plant conditions or phenomena. The dependence between operator actions is included in the uncertainty of the Level 2 split fractions."	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3148	Unresolved	LE-F1b	RG1.200 Peer Review F&O LE-F1b-01, Finding There is no documented evidence that ANO-2 compared their LERF results to the results of other similar plants to confirm the reasonableness of the results with respect to relative contribution and frequency and ranking of contributors.		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
A2-3149	Resolved	MU-B4	RG1.200 Peer Review F&O MU-B4-01, Finding There is no reference to a peer review for upgrades. Procedure EN-DC-151, PSA Maintenance and Update, could be revised to include the requirement for a peer review when the PSA is upgraded. It should be noted that a PSA update does not require a peer review. An upgrade could include the following: change in methodology, change in software, or any other change that could be defined as an upgrade.	Section 5.5 (Reference 7) and Attachment 9.2, Item 6, in Nuclear Management Manual procedure EN-DC-151, address the check for a PSA upgrade per the ASME/ANS PRA Standard.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-5859	Unresolved	IFEV-A7	Screening for scenarios was performed within PSA-ANO2-01-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. 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-ANO2-01-IF-IE), the screening criteria does not meet the numerical threshold of IE-C6. Maintenance event screening is currently documented as a ~1E-05/yr frequency. Even with the proposed 0.1 factor to account for maintenance that breaches the pressure boundary, the criteria is still not met. (This F&O originated from SR IFEV-A7), Finding F&O	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
A2-5860	Unresolved	IFQU-A5	The Focused-Scope Internal Flooding PRA Peer Review identified in Fact and Observation (F&O) 19-5 identified the following. The calculation of HEPs is documented in the quantification notebook (PSA-ANO2-01-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: HR-E3, HR-E4, and HR-G5: Operator walkthoughs or talkthroughs of the flooding isolation actions were not performed. 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. HR-G6: A consistency check of the HEPs was not performed. 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. (This F&O originated from SR IFQU-A5), Finding F&O	Additional discussion will be included in the IFA documentation and adjustments to some flooding HFE timing will be implemented.	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.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A2-5861	Unresolved	IFQU-A5	The Focused-Scope Internal Flooding PRA Peer Review identified in Fact and Observation (F&O) 19-7 identified the following: Flood specific post-initiator HFEs were developed to support the internal flood quantification. These HFEs were documented in PSA-ANO2-01-IF-QU Attachment 1. The industry standard HRA calculator was used to perform the analysis. 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. (This F&O originated from SR IFQU-A5), Finding F&O	This is a documentation issue and additional discussion will be included in the IFA documentation.	This is a documentation issue. 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.
A2-5862	Unresolved	IFQU-A8	The Focused-Scope Internal Flooding PRA Peer Review identified in Fact and Observation (F&O) 19-11 identified the following: The internal events PRA model, updated by FRANX to include flooding impacts, properly considers both flooding impacts and random equipment failures and human errors. However, at least one instance was identified in which a modified internal events HEP was not properly included in the integrated model. See event MFW2XHE-FO-MFWIS, which was supposed to be set to 1.0 in all auxiliary building and turbine building flooding scenarios that occurred at elevation 329 feet and higher. (This F&O originated from SR IFQU-A8), Finding F&O	The impacted HEPs will be updated as needed in the IFA model and documentation.	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.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
A2-5863	Unresolved	IFPP-B3	The Focused-Scope Internal Flooding PRA Peer Review identified in Fact and Observation (F&O) 20-4 identified the following: No documentation of modeling uncertainties is provided in any of the internal flooding notebooks. (This F&O originated from SR IFPP-B3), Finding F&O	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.
A2-5864	Unresolved	IFQU-A7	The Focused-Scope Internal Flooding PRA Peer Review identified in Fact and Observation (F&O) 20-7 identified the following: 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. (This F&O originated from SR IFQU-A7), Finding F&O	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.
A2-5865	Unresolved	IFQU-A7	The Focused-Scope Internal Flooding PRA Peer Review identified in Fact and Observation (F&O) 20-8 identified the following: A parametric uncertainty analysis was not performed as required by QU-E1. (This F&O originated from SR IFQU-A7), Finding F&O	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
A2-5866	Unresolved	IFQU-A7	The Focused-Scope Internal Flooding PRA Peer Review identified in Fact and Observation (F&O) 20-9 identified the following: 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. (This F&O originated from SR IFQU-A7)	The impacted HFEs will be updated as needed in the IFA model and documentation.	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.
A2-5867	Unresolved	IFQU-A7	The Focused-Scope Internal Flooding PRA Peer Review identified in Fact and Observation (F&O) 20-10 identified the following: 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. (This F&O originated from SR IFQU-A7), Finding F&O	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
A2-5868	Unresolved	IFQU-A5	The Focused-Scope Internal Flooding PRA Peer Review identified in Fact and Observation (F&O) 20-11 identified the following: Flood specific post-initiator HFEs were developed to support the internal flood quantification. These HFEs were documented in PSA-ANO2-01-IF-QU Attachment 1. The industry standard HRA calculator was used to perform the analysis. 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. (This F&O originated from SR IFQU-A5), Finding F&O	This is a documentation issue and additional discussion will be included in the IFA documentation.	This is a documentation issue and additional software testing is not expected to impact the internal flooding results or STI evaluations performed in accordance with the SFCP.
A2-5869	Unresolved	IFQU-A10	The Focused-Scope Internal Flooding PRA Peer Review identified in Fact and Observation (F&O) 20-12 identified the following: LERF was not considered in the internal flooding PRA. (This F&O originated from SR IFQU-A10), Finding F&O	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
N/A	Resolved	LE-C2b	This is not a finding but an observation of this SR meeting CC-1 with no associated findings. Repair was not addressed.	The observation noted pertained to a prior version of the ANO-2 LERF analysis dated February 2010. The ANO-2 LERF analysis has since been completed was most recently updated in June 2015, and is undergoing current update. The current analysis for the LERF results is assumed conservative with respect to potential for restoration of mitigation equipment. No material impact is expected for this issue.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP. Minimal or no impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.
N/A	Resolved	LE-C8a	This is not a finding but an observation of this SR meeting CC-1 with no associated findings. For ANO-2, the systems with components inside containment and thus subject to the severe environment are the containment spray and fan coolers. ANO-2 does not credit sprays and fan coolers for averting LERF. The credited human actions are performed outside containment.	The observation noted pertained to a prior version of the ANO-2 LERF analysis dated February 2010. The ANO-2 LERF analysis was most recently updated in June 2015, and is undergoing current update, where the ANO-2 LERF analysis has since been completed and additional documentation is provided to address this issue.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP. This is a documentation issue. Minimal or no impact on STI evaluations.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
N/A	Resolved	LE-C9a	This is not a finding but an observation of this SR meeting CC-1 with no associated findings. A review of Figures 4.1-1 and 4.1-2 of PRA-A2-01-003S12 indicates that ANO-2 does not appear to model equipment operation post-containment failure.	The observation noted pertained to a prior version of the ANO-2 LERF analysis dated February 2010. The ANO-2 LERF analysis was most recently updated in June 2015, and is undergoing current update, where the ANO-2 LERF analysis has since been completed and additional documentation is provided to address this issue.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP. This is a documentation issue. Minimal or no impact on STI evaluations.
N/A	Resolved	LE-C9b	This is not a finding but an observation of this SR meeting CC-1 with no associated findings. A review of Figures 4.1-1 and 4.1-2 of PRA-A2-01-003S12 indicates that ANO-2 does not appear to model equipment operation post-containment failure.	The observation noted pertained to a prior version of the ANO-2 LERF analysis dated February 2010. The ANO-2 LERF analysis was most recently updated in June 2015, and is undergoing current update, where the ANO-2 LERF analysis has since been completed and additional documentation is provided to address this issue.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP. This is a documentation issue. Minimal or no impact on STI evaluations.

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Importance to Application
N/A	Resolved	LE-C10	This is not a finding but an observation of this SR meeting CC-1 with no associated findings. ANO-2 does not take credit for scrubbing for containment bypass events.	The observation noted pertained to a prior version of the ANO-2 LERF analysis dated February 2010. The ANO-2 LERF analysis was most recently updated in June 2015, and is undergoing current update, where the current analysis for the LERF results is assumed conservative with respect to potential for scrubbing for bypass sequences.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP. Minimal or no impact on the results. However, impact of this finding is expected to be assessed in case-by-case STI evaluation.

3.3 ANO-2 Fire PRA Model

3.3.1 Plant Changes Not Yet Incorporated

Similar to the internal events 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 open items (peer review Findings for ASME/ANS PRA Standard Capability 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. All ECs related to the implementation of NFPA-805 were implemented prior to restart of the last ANO-2 refueling outage. The current ANO-2 self-approval fire PRA model captures these modifications. 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-2 fire PRA model has undergone several peer reviews, including a full scope and an HRA focused-scope peer review. Additionally, two focused scope peer reviews were conducted that focused on fire modeling. 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-2 fire PRA model:

- In September 2009, a Westinghouse Owners Group peer review was conducted under LTR-RAM-II-09-046 (Reference 9). There was a total of 64 F&Os, which included 30 suggestions, 33 findings, and one best practice. The conclusion of the review was that the ANO-2 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 the NUREG/CR-6850 methodologies correctly.

- In October 2011, a focused-scope peer review was conducted by the Westinghouse Owners Group. The peer review was performed to include High Level Requirements (HLRs) FSS-A, C, D, E, and H with respect to the ANO-2 transient fire scenario development process. Secondarily, Entergy also requested review of the treatment of “Closed but Unsealed” electrical cabinets and the development of THIEF-Informed Zones-of-Influence (ZOI). This focused-scope peer review is documented in LTR-RAM-II-11-064 (Reference 10). There was a total of five F&Os identified, which included two suggestions and three findings. The Westinghouse Owners Group concluded that the ANO-2 fire PRA is consistent with the ASME/ANS PRA Standard and supported risk-informed applications.

- In June 2014, a focused-scope peer review was conducted by Scientech (Reference 11), which concentrated on the HRA portion of the Fire PRA and was performed against RG 1.200, Revision 2. There was a total of four F&Os, which included two suggestions and two findings. The peer review found that the ANO-2 fire HRA was performed consistent with the guidance set forth in NUREG/CR-1921, and specifically with NUREG/CR-1921, Attachment B, “Detailed Quantification of Fire Human Failure Events Using the EPRI Fire HRA Methodology.”

- In November 2012, a focused-scope peer review was conducted by Mardy Kazarians of Kazarians & Associates, Inc, (Reference 19), 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. There were a total of five F&Os identified, which included five suggestions and no findings.

- In September 2016, a focused-scope peer review was conducted by Dr. Jason Floyd of JENSEN HUGHES (Reference 20), which concentrated on the fire modeling using the Fire Dynamics Simulator (FDS) covering requirements FSS-C, D, and H. No F&Os were identified in this focus-scope peer review.

3.3.3 Consistency with Applicable PRA Standards

As discussed in Section 3.1, the ANO-2 fire PRA model was updated in 2017. 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) Capability Category II of all SRs. Current Entergy PRA documentation includes an individual self-assessment that documents how each HLR and SR are met. A gap related to documentation from this individual self-assessment was identified and documented in MCR A2-5585 (missing table from Success Criteria, Accident Sequence, and System Analysis packages). This documentation issue is expected to be resolved as part of the revision 6p00 model update.

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-2 Fire PRA Self Approval Model and discussed in Table 2, 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”

- NUREG-2169, Rev. 0, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009, January 2015”

- NUREG-2178, Rev. 0, “Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLEFIRE) Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume, April, 2016”

The latest full scope peer review for ANO-2 fire PRA model was conducted in September 2009 using RG 1.200, Revision 2. In addition, focused scope peer reviews were performed in 2011, 2012, 2014, and 2016 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 these peer reviews were addressed properly. Table 2 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. Table 3 also lists SRs associated with the fire PRA; however, 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 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 Capability 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 no open items and no sensitivity case is required.

Table 2

List of Finding F&Os against the ANO-2 Fire PRA Model

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	CF-A1	Fire PRA Peer Review Finding CF-A1-01 The FRANC AlteredEvents Table, Appendix F in the ANO-2 Scenario Report (ERIN doc. # 0247-06-0006.05 Rev. 0) appears to have incorrect probabilities in the following instances: 1) Events related to the letdown flow control valves 2CV4816 & 2CV4817 apparently used the wrong probability from NUREG/CR-6850 Table 10-2 for fire zones 2098-C & 2108-S. In these fire zones, database xx2.mdb shows that the applicable cables (2I016Q and 2I016N/P, respectively) are routed through trays instead of conduits as apparently assumed in Appendix F. NUREG/CR-6850 Table 10-2 allows probabilities in the range of 0.02 to 0.1 for inter-cable shorts in trays rather than the 0.01 cited in Appendix F which would be acceptable if the cables were in conduits. 2) Appendix F states the 0.0006 probability used for event YMP2P35BFF in fire zones 2100-Z and 2109-U-C comes from the HFE AHF2CSAS1P. In the HRA toolbox spreadsheet hfe_cp New Fire PRA HFEs (4-23-09).xlsm provided shows that the HEP for this HFE is actually 3.2E-4.	The altered events table was revised to correct, and provide a summary of the basis for, the assigned probabilities. 1) Spurious actuation of these valves requires two, concurrent, proper polarity shorts to another 4-20mA cable. Table 10-2 of NUREG/CR-6850 provides a probability range of 0.02 to 0.1 for an inter-cable short from one multi-conductor cable to another in a tray for a DC circuit. Assuming an average probability of 0.06 for each short, the overall probability is 0.06 * 0.06 or 0.0036. This discussion is documented in Section 12 of the Fire Scenarios Report (PRA-A2-05-003). 2) YMP2P35BF was originally included in the model as an MSO for depletion of the RWT. However, it has since been removed due to the length of time required to deplete the RWT, the operator cues for rapid RWT depletion, reactor building (RB) Spray pump operation, and the alignment to the sump, if required. Therefore, this event is removed from the Altered Events list.	1) The conditional probabilities of spurious hot shorts have been updated using data from NUREG/CR-7150, Vol.2 , “Joint Assessment of Cable Damage and Quantification of Effects from Fire” . The disposition of this item in the F&O is remains valid 2) 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 to Reflect New Guidelines	Importance to Application
A2-3736 *continued*	Resolved	CF-A1 *continued*	3) Appendix F states the 0.22 probability used for event EB12A3XXXF in fire zone 2100-Z comes from the HFE EHF2A309XP. In the HRA toolbox spreadsheet hfe_cp New Fire PRA HFEs (4-23-09).xlsm provided shows that the HEP for this HFE is actually 2.2E-2. Use of incorrect probabilities in the FRANC AlteredEvents Table would introduce errors in results obtained using FRANC. Correct errors in probabilities used or document justification for values currently found in the FRANC AlteredEvents Table.	3) EHF2A309XP has been added to the fault tree. Therefore, the altered event for EB12A3XXXF has been removed.	3) The HRA methodology used in determining the probability of failure for significant operator actions in the Fire PRA was revised to follow the guidance of NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines. The use of NUREG-1921 does not affect the original disposition to this F&O.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3736	Resolved	CF-B1 Other Affected SRs FSS-A3, HRA-B1	Fire PRA Peer Review Finding CF-B1-01 The FRANC AlteredEvents Table, Appendix F, in the ANO2 Scenario Report (ERIN doc. # 0247-06-0006.05 Rev. 0) makes recurring reference to ANO Engineering Change EC13540, "ANO2 CABLE ROUTING EXCLUSIONS TO SUPPORT FIRE PRA FOR NFPA-805" to document that cables associated with a given event do not go through the cited fire zone. A review of EC13540 shows that it does identify the component and associated cables, but documents different fire zones than those listed in Appendix F.	The altered events table in Attachment F of the Fire Scenarios Report (PRA-A2-05-003) has been revised so that the reference to EC13540, "ANO-2 Cable Routing Exclusions to Support Fire PRA for NFPA-805," has been removed for the items mentioned in the finding. Where EC 13540 is referenced, the events can be tied to the EC directly. Also see Entergy response to NRC PRA RAI 01, dated November 7, 2013 (2CAN111301) (ML13312A877).	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 to Reflect New Guidelines	Importance to Application
A2-3736 *continued*	Resolved	CF-B1 Other Affected SRs FSS-A3, HRA-B1 *continued*	For example, events QAV200798C and QSV200798C for valve 2CV-0798-1 are assigned a probability of zero because the cables for that valve do not pass through fire zones 2007-LL and 2024-JJ. EC13540 shows that cables for valve 2CV 0798-1 do not pass through fire zone 2101-AA without any mention of fire zones 2007-LL and 2024-JJ. A review of PDMS data shown in database file xx2.mdb confirm that 2CV-0798-1 cables do not pass through fire zones 2007-LL and 2024-JJ, but EC13540 cannot be used as a reference to support that conclusion. This is applicable to almost all of the events set to zero with reference to EC13540. In three instances, an event was incorrectly set to zero based on EC13540 when cables associated with the component actually do pass through the cited fire zone based on PDMS information found in database xx2.mdb: 1) Event QSV200798R for valve 2SV-0798-1 has an associated cable R2D2301A identified in EC13540 passing through fire zone 2099-W in tray EC122. 2) Event PMV210401R for valve 2CV-1040-1 has an associated cable R2B53B1E identified in EC13540 passing through fire zone 2200-MM in tray EC114. 3) Event ECB2A409XD for breaker 2A409 has associated cables 2A409A thru H and J thru M identified in EC13540 passing through fire zone 2200-MM in trays DA020, 030 thru 035, 037 & 039.			

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
A2-3736 *continued*	Resolved	CF-B1 Other Affected SRs FSS-A3, HRA-B1 *continued*	Event ECB2A409XR for breaker 2A409 has a probability of 0.1 assigned for fire zone 2200-MM-B while two other events for the same breaker in that fire zone are assigned probabilities of 0. Note that EC13540 does not address this fire zone at all. Inadequate justification provided for setting numerous event probabilities to zero and in three instances event probabilities inappropriately set equal to zero. Use of incorrect probabilities in the FRANC AlteredEvents Table would introduce errors in results obtained using FRANC. Revise EC13540 to address the fire zones listed in the FRANC AlteredEvents Table and fix probabilities where re-analysis of cable routing shows that the cables actually do pass through the cited fire zones.			
A2-3736	Resolved	CS-A7 Other Affected SRs ES-B1, FQ-D1	Fire PRA Peer Review Finding CS-A7-01 Although cables are considered for one containment bypass event (MOV 2CV-3200-2 TRANSFERS OPEN), there are no other containment isolation failures considered in the LERF model. The containment purge system is not considered (it is screened from the internal events model) and other containment bypass paths may exist, but are not identified. There is currently an open internal event PRA F&O (LE-D6-01) on the containment isolation system, expressing no confidence that the analysis represents a realistic assessment.	A review of the containment piping and venting paths were reviewed for impact on large early release. No LERF paths were identified for the internal events. Only two potential LERF paths were identified due to multiple spurious operations of the isolation valves: the containment purge supply and return lines (penetrations 2V-1 and 2V-2). However, each of these lines includes two AOVs and an MOV for isolation. The valves are in series with different power supplies (i.e, red and green powered), have independent circuits, and power supplies that are separated by fire zone with the exception of the control room.	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 to Reflect New Guidelines	Importance to Application
A2-3736 *continued*	Resolved	CS-A7 Other Affected SRs ES-B1, FQ-D1 *continued*	Cable and circuit failure modes affecting containment bypass are not completely considered or dispositioned in the Fire PRA plant response model for LERF. Review potential containment bypass pathways beyond those in the current internal events PRA, including containment purge, that could lead to a LERF.	The controls within the control room divided between two cabinets (2C16 and 2C17). These cabinets have sufficient separation and barriers to prevent a fire from spreading from one cabinet to another.	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.
A2-3736	Resolved	CS-A8 Other Affected SR FSS-C5	Fire PRA Peer Review Finding CS-A8-01 ER-ANO-2003-0450-000 Rev 0 is credited in the self-review as determining that only 1% of cables at ANO-2 are thermoplastic and therefore of minimal significance. This ANO-generated document is not sufficiently complete to justify this conclusion. This engineering reply document states that 380 thermoplastic cables were identified in PDMS, but listed only 10 specific cables as being in "safety significant fire zones". A brief review of safe shutdown components conducted with ANO engineering personnel identified a safe shutdown neutron monitoring system as having thermoplastic cables which is not mentioned in the engineering reply document cited above. A complete plant-wide review of thermoplastic cable used for safe-shutdown components should be performed, documented and referenced in the Fire PRA Component and Cable Selection Report. SR requirement to include treatment of thermoplastic cable failure was not satisfied.	CALC-ANOC-FP-09-00019, Rev. 0, EC-6964, "Safe Shutdown Cable Jacket Insulation Types at ANO," evaluates the cables at both units and concludes that of the over 4600 cables reviewed, less that 0.3% have thermoplastic insulation. This calculation also confirms that thermoplastic cables are not used in power supply circuits. Thus, thermoplastic insulation at ANO is of minimal significance. CALC-ANOC-FP-09-0019 is referenced in the Fire PRA Scenarios Report. The resolution of ANO-2 CS-A8-01 was reviewed and accepted by the ANO-1 Peer Review Team.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	CS-A9 Other Affected CS-A5	Fire PRA Peer Review Finding CS-A9-01 The ANO Fire PRA guidance was contradictory with respect to cable selection with respect to proper polarity hot shorts on ungrounded dc circuits, and it was unclear whether these hot shorts were appropriately considered. "Safe Shutdown Cable Analysis," Calculation Number 85-E-0087-24, Revision 1 was designated as the reference for the scope. Section 4.2.5 states: "For ungrounded DC circuits, two hot shorts of the proper polarity (without grounding) causing spurious operation is not considered credible except for high-low pressure interface components." However, the ANO-2 self-assessment for CS-A9 states: "Postulated proper polarity hot shorts on ungrounded DC circuits (e.g., RCS head vents) were considered as documented in the PDMS database." Further, in CALC-85-E0087-24, Safe Shutdown Cable Analysis, conflicting criteria was identified in Section 4.2 of the calculation. Criteria 4.2.2 states that all DC grounded and ungrounded circuits must consider any and all shorts, hot shorts, shorts-to ground and open circuits. Criteria 4.2.3 states that all ungrounded circuits (both AC and DC) will be analyzed as if the circuit is grounded to account for the possibility of experiencing a ground fault. However, criteria 4.2.5 then states that for ungrounded DC circuits, two hot shorts of proper polarity (without grounding) are not considered credible except for high-low pressure interface components.	The F&O indicates that Criteria 4.2.2, 4.2.3, and 4.2.5 in CALC-85-E-0087-24 are contradictory and that this guidance for cable analysis should be changed. However, the criteria in CALC-85-E-0087-24 come from NUREG-1778 and Generic Letter 86-10, and provide a consistent approach to circuit analysis. This approach keeps the analyst from making false assumptions based upon a “single failure” safety analysis instead of more appropriately considering multiple concurrent failures expected as the result of a fire. The criteria in question are concerned with DC circuit analysis and ungrounded AC circuits which are similar to ungrounded DC in how they respond to circuit failures. The 125 VDC power and control circuits at ANO are ungrounded and typically only shields in low voltage DC instrumentation loops are grounded. Not grounding DC systems allows a single fault on either the positive or negative side of the circuit to occur without jeopardizing the function of the DC system. The criteria used for ungrounded systems provide a methodical approach as discussed below. Criterion 4.2.2 of CALC-85-E-0087-24 is a definition of the scope of DC circuits considered and prevents the exclusion of any circuit associated with safe shutdown equipment without engineering review/analysis.	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 to Reflect New Guidelines	Importance to Application
A2-3736 *continued*	Resolved	CS-A9 Other Affected CS-A5 *continued*	Criteria 4.2.5 conflicts with 4.2.2 and 4.2.3 for several reasons: (a) 4.2.2 criterion requires that any and all hot shorts be considered, and (b) 4.2.3 criterion requires that ungrounded dc circuits be grounded, and it would only take one hot short of proper polarity from the same DC source to spuriously operate the high/low pressure interface. These criteria need to be aligned with each other. Additional contradictory evidence is found in the "NFPA-805 Fire PRA Modeling of Multiple Spurious Operations (MSO)" document CALC-ANO2-FP-09-00016, Rev. 0. For example, the disposition of PWR MSO 10 states that inter-cable hot shorts are not postulated due to the use of thermoset cables. The guidance and discussion of proper polarity hot shorts on ungrounded dc circuits is contradictory and the correct disposition of this requirement could not be determined. Clarify the position on proper polarity hot shorts in ungrounded dc circuits and document that the ANO-2 Fire PRA includes consideration of proper polarity hot shorts on ungrounded dc circuits, besides high-low interface components.	Criterion 4.2.3 of CALC-85-E-0087-24 is a simplified methodology to DC circuit analysis. Since safe shutdown deals with multiple failures it is possible to have failure in a separate circuit create the initial ground fault. Requiring that all ungrounded circuits (both AC and DC) be analyzed as if the circuit is grounded simplifies the analysis to determine circuit failure. A single short/fault in the circuit being reviewed can result in failure since a ground is already assumed to be established. This criterion requires all circuits, grounded and ungrounded, to be evaluated in the same manner. Criterion 4.2.5 of CALC-85-E-0087-24 will allow an exclusion of a specific circumstance where a cable in the subject circuit faults to a second cable (two hot shorts of proper polarity). This unique intercable short requires that it be of the appropriate DC voltage, proper polarity (positive to positive/negative to negative), occurs exclusive of any fire induced grounds, and not be a high-low pressure boundary. This criterion is the DC equivalent of Criterion 4.2.4 which is for 3-phase AC.		

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
A2-3736 *continued*	Resolved	CS-A9 Other Affected CS-A5 *continued*		To summarize, Criterion 4.2.2 provides a baseline, Criterion 4.2.3 a simplified methodology, and Criterion 4.2.5 is an exclusion for a specific set of circumstances. Therefore, the criteria are consistent with each other and no revision to CALC-85-E-0087-24 is necessary. Also, the F&O indicates that additional contradictory evidence is found in the "NFPA-805 Fire PRA Modeling of Multiple Spurious Operations (MSO)" document CALC-ANO2-FP-09-00016, Rev. 0. For example, the disposition of PWR MSO 10 states that inter-cable hot shorts are not postulated due to the use of thermoset cables. This inaccurate wording has been removed from CALC-ANO2-FP-09-00016. Inter-cable hot shorts are postulated in the fire PRA model and NSCA as described in the MSO report.		

MCR Number	Status	Applicable SR(s)	Finding/Observation	Disposition	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	CS-C4	Fire PRA Peer Review Finding CS-C4-01 Electrical distribution system over-current coordination and protection analysis was performed using the methodology of ANO Upper Level Document "Electrical Protection/ Coordination", ULD-0-TOP-12 Revision 3, 11/25/2002. This upper level document references the ANO-2 plant specific analysis found in engineering calculation 84-E-0103-01, Revision 8, "General Criteria for Safety Bus", 01/21/2000. No references or discussion of this review is provided in the fire PRA documentation. Provide in the appropriate portion(s) of the fire PRA documentation references to the review of electrical distribution system over-current coordination and protection analysis. References to the review of electrical distribution system over-current coordination and protection analysis are required in Fire PRA documentation to satisfy this SR. Provide in the appropriate portion(s) of the Fire PRA documentation references to the review of electrical distribution system over-current coordination and protection analysis.	Section 4.4 of the Component and Cable Selection Report, 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 ULD-0-TOP-12, Upper Level Document ANO-1 and ANO-2 Electrical Protection/ Coordination, Rev. 3. Also see Entergy response to NRC PRA RAI 01, dated November 7, 2013 (2CAN111301) (ML13312A877).	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	ES-C1	Fire PRA Peer Review Finding ES-C1-01 Task 12, Post Fire HRA was reviewed for detailed analysis performed for credited operator actions. For many of the recovery actions control room instrumentation and annunciators were credited as cues. In many cases, these instruments and annunciators (along with supporting cables and power supplies) were not added to the equipment list and validated for function. In some cases, the cues stated may not be necessary since it is expected that the operator takes the manual actions regardless of cues stated. In other cases, the operators would not know the actions are required unless the cues were available. Example of instrumentation needed: RHGISORCPBO (Operators Fail to Isolate RCP Bleed-off) credits RCP bleed off temperature from annunciator 2K11 for knowledge that spurious operation has occurred. Example of instrumentation (annunciators) credited, though may not be needed include EHF2A309XP (Operator fails to locally close 2A-309 to restore off-site power to 2A-3). This action credits annunciators for low bus voltage and EDG (Emergency Diesel Generator) trouble. However, it appears that the step to verify 2A1 energized from Off Site power would be followed if verification could not take place due to loss of instrumentation. For each HFE, check credited instrumentation and add as appropriate to the equipment list. Ensure satisfaction of ES-B4 such that supporting power supplies are included in equipment/cable selection.	At the time of the peer review, most of the Post Fire HRAs included information related to the instrumentation for operator cues in the Fire Scenarios report. Since the peer review, the HRA information was moved to a separate report (PRA-A2-05-007, Rev. 0). The HRA Notebook, Section 4.2 describes the correlation of instrumentation for HFEs to Appendix R instruments that have been confirmed to be available following a fire. The existing HFEs were evaluated to determine if non-Appendix R instrumentation is required. The additional detail on the HRA instrumentation had a minor impact on HRA probabilities and HRA credit in the Fire Scenarios. The addition of detail for the HRAs represents an enhancement of the justification for HRA credit in the fire scenarios and does not impact the Fire PRA methodology. PRA-A2-05-002, ANO2 Fire PRA New Human Failure Events, was revised to indicate the appropriate operator cues. Event RHGISORCPBO was renamed to RHF2BLOFFP to reflect the ANO basic event naming convention, however, this event is not currently credited in the Fire PRA model. EHF2A309XP has sufficient cues to ensure operator actions are taken; additional instrumentation is not necessary.	The HRA methodology used in determining the probability of failure for significant operator actions in the Fire PRA was revised to follow the guidance of NUREG 1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines. The use of NUREG 1921 does not affect the original disposition to this F&O.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	ES-C1	Fire PRA Peer Review Finding ES-C1-02 Appendix R instrumentation for Steam Generator pressure and level were credited for manual operation of AFW based upon adequate instrumentation demonstrated by the Appendix R analysis. The Appendix R analysis only validated that a minimum of one train was available. Since the PRA model may credit other trains, validation that adequate instrumentation for that train is available to support manual operations was not performed. Evaluate instrumentation availability either within the PRM or through separate evaluation and document.	As stated in the F&O, one train of level instrumentation will be available to the operators for manual operation of the AFW. Also, additional level and pressure indication is to be installed at a proposed AFW control and instrumentation panel outside the control room. The culmination of these instruments will ensure sufficient cues for all fire scenarios. In addition, the operator cues for manual start of the AFW pump is based on both SG instrumentation and failure of the MFW and EFW pumps to operate. Since the SG instrumentation is not the only cue used for this operator action and no fire scenario damages all SG instrumentation, the HEP for manual start of AFW is not impacted by a failure of one train of Appendix R instrumentation.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	ES-D1 Other Affected SR HRA-B3	Fire PRA Peer Review Finding ES-D1-01 ANO Fire Scenarios Report, 0247-06-0006.05, Appendix E has numerous documentation issues: 1) Some Basic Events (e.g. AHF2CSAS1P) do not have entries under "Operator Cues and Components Credited with Providing Operator Cues for Pose Fire Safe Shutdown" column, even though the "Required Instrumentation Available" column is marked Yes. 2) Column titles contain references to footnotes, but the footnotes are not in the document 3) The instrumentation identified as required for operator actions are listed in noun format, not instrumentation identifiers (e.g. "Steam generators pressure indications") 4) No explanation is given regarding whether or not all instrumentation listed in column for operator cues & components 5) No link to documentation is provided to basis of conclusion in "Required Instrumentation Available" column Since there is information missing from Appendix E that is necessary for determining whether or not SR ES-C1 is met, this is a documentation finding. It also has an impact on the determination of SR HRA-B3.	At the time of the peer review, most of the Post Fire HRAs included information related to the instrumentation for operator cues in the Fire Scenarios report. Since the peer review, the HRA information was moved to a separate report (PRA-A2-05-007, Rev. 0). 1) The column "Operator Cues and Components Credited with Providing Operator Cues for Pose Fire Safe Shutdown" has been populated for all credited HFEs. 2) The footnotes associated with this Appendix have been moved to the Assumptions section of the HRA report. These assumptions address the justification of Fire HRA multipliers and the definitions of accessible and simple actions to account for fire impacts on HRAs. 3) The instruments for each operator action used in the Fire PRA are provided using the component identifiers in the Cue Instruments / Components column. 4) The Cue Instruments / Components column has been added to the table in Appendix A to provide the instruments used.	The HRA methodology used in determining the probability of failure for significant operator actions in the Fire PRA was revised to reflect the requirements of NUREG-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines. The use of NUREG 1921 does not affect the original disposition to this F&O.	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 to Reflect New Guidelines	Importance to Application
A2-3736 *continued*	Resolved	ES-D1 Other Affected SR HRA-B3 *continued*	Revise Appendix E of the report to include missing information and to clearly explain the basis of conclusion of availability of a component; this could be achieved by adding a column to reference the Safe Shutdown Capability Assessment or Safe Shutdown Equipment List - whichever document is more relevant.	5) The availability of at least one train of Appendix R instrumentation has been verified through the deterministic safe shutdown analysis. The availability of non-Appendix R instruments is documented based on discussions with Operations.		
A2-3736	Resolved	ES-D1 Other Affected SRs ES-C2, HRA-A3	Fire PRA Peer Review Finding ES-D1-02 Attachment C of the Fire Scenario Report 0247-06-0006.05 is the Fire PRA Simulator Review, which assesses the instrumentation needs of the operators during / following a fire event. The review was conducted by a simulator operator and a Senior Reactor Operator and is formatted such that all of the appropriate questions are addressed to ensure that the SR is met. However, the basis for the conclusions is not included in the report. The condition for meeting the Category II requirement is answered by the questions, "Do the operators only rely on the specifically identified instruments?" and "Which tiles cause the operator to take and immediate action?" The answers are that they do not rely on only one indicator and that there are no immediate actions taken due to annunciation. There is no other supporting documentation to demonstrate that these responses are correct.	The HRA Notebook, PRA-A2-05-007 was revised to include the information discussed in this F&O. Appendix A includes all of the instruments that would provide indications for the operator actions. In addition, OP-2203.049 verifies the Appendix R instruments that are available for the worst case fire in each fire area. The operators are trained to rely on these instruments for cues to perform the operator actions. COPD-001, “Operations Expectations and Standards” Section 5.4.2 provides the following direction; “Numerous events within the industry have occurred because the Operator performing the reactivity change only focused on one parameter to determine core response, such as, only monitoring RCS average temperature following a reactivity addition and ignoring other critical parameters. This practice must be avoided.	The HRA methodology used in determining the probability of failure for significant operator actions in the Fire PRA was revised to follow the guidance of NUREG 1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines. The use of NUREG 1921 does not affect the original disposition to this F&O.	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 to Reflect New Guidelines	Importance to Application
A2-3736 *continued*	Resolved	ES-D1 Other Affected SRs ES-C2, HRA-A3 *continued*	The ANO team confirmed that the analysis had verified the responses, but there is no documentation of the basis of the conclusions; therefore, it is very difficult to recreate the same conclusions. Since critical documentation is missing, this is considered to be a finding as opposed to a suggestion. Revise Attachment C of the report to include either a discussion regarding the basis for the answers or at a minimum to reference the applicable document. Also, clarify the relevancy of the unanswered questions or provide a response.	The individual as well as the Operations team must use diverse and multiple indications to monitor changes to the core as a result of reactivity manipulations." Therefore, a basis is available to support the conclusions of the simulator review.		
A2-3736	Resolved	FSS-A4	Fire PRA Peer Review Finding FSS-A4-01 Plant drawing E-2872 was reviewed to identify some sample targets (EC3021, EC3022, B4124, B4125, and J4519) within the zone of influence of 2D-35 (Scenario 2109-U-B). All the targets were listed in Attachment A, except B4124. A review and plant walk-down should be performed to determine if B4124 should be added to the list of targets in Attachment A of ERIN Report 0247-06-0006.05. Because this is a conduit, the location of conduits may not necessarily match the location in the drawing. If this conduit is located in the zone of influence of 2D-35, it will need to be listed as a target. Verify PDMS information is consistent with walk-down results and drawings.	Conduit B4124 was determined to be within the zone of influence of 2D-35. Therefore, conduit B4124 was added to the list of targets for scenario 2109-U-B in Attachment A of the Scenario Report. This finding is associated with a specific scenario with a missing target. A review of drawings identified targets that were difficult to locate physically in the field. This review was performed for all scenarios in congested fire zones. The walkdown target list includes notes that identify some of the targets that were identified from the drawing review versus the original. Also see Entergy response to NRC PRA RAI 01, dated November 7, 2013 (2CAN111301) (ML13312A877).	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	FSS-A5	Fire PRA Peer Review Finding FSS-A5-01 The ignition frequency used in scenario 2111-T-B is for the electrical panels and not a battery charger. The CDF quantification for this scenario will need to be revised to use the correct ignition frequency. A related F&O FSS-D10-01 identified that a battery charger was missed from the ignition source walk-down sheets and the CDF quantification results were not correct.	The CDF quantification for 2111-T-B has been revised to use the appropriate ignition frequency. See PRA-A2-05-003 - the ANO-2 Fire Scenarios Report for details. The ignition source data sheet and walkdown sheet for compartment 2111-T was revised in CALC-08-E-0016-01 to include battery charger 2D-33.	The ignition frequencies have been revised to encompass NUREG 2169 Rev. 0, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009, January 2015.” The use of this more recent information does not affect the original disposition to this F&O.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.
A2-3736	Resolved	FSS-B1	Fire PRA Peer Review Finding FSS-B1-02 The glass partition between the two control room fire compartments (129-F and 2199-G) is not a fire rated barrier. The calculation of the control room abandonment CDF for Unit 2 only considers the fire initiating frequency for Unit 2, and does not consider a fire in the Unit 1 control room. The fire ignition frequency for the Unit 2 control room abandonment is under estimated. Include the fire ignition frequency for a Unit 1 control room fire in the Unit 2 control room abandonment CDF.	The *Evaluation of Unit 2 Control Room Abandonment Times at the Arkansas Nuclear One Facility* [ANO2-FP-09-00013] defines the conditions that are assumed to cause control room abandonment and reliance upon alternate shutdown actions. Abandonment times are assessed for ANO-2 for fire scenarios originating in the ANO-1 control room and the sensitivity study also assumes failure of the glass barrier separating the ANO-1 and ANO-2 control rooms.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	FSS-B2	Fire PRA Peer Review Finding FSS-B2-01 The control room abandonment (CRA) scenario for Unit 2 (2199-G-B) assumes a CCDP of 0.1 for an alternate shutdown scenario. The basis for this CCDP in the Fire Scenarios Report states that this is "a conservative CCDP, probability of failure of shutdown from outside the control room, given that a deterministic analysis has been performed and feasibility of required actions have been assessed to ensure feasibility of this shutdown scenario for a worst case fire". This is not a sufficient basis to validate a CCDP value of 0.1. A determination of a bounding risk for the MCR abandonment scenario is not demonstrated with the assumed CCDP.	A detailed Control Room Abandonment analysis [CALC-09-E-0008-10] was performed to calculate the CCDP for MCR Abandonment. Also see Entergy response to NRC PRA RAI 01, dated November 7, 2013 (2CAN111301) (ML13312A877).	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.
A2-3736	Resolved	FSS-C5	Fire PRA Peer Review Finding FSS-C5-01 Per ER-ANO-2003-0450-000, Revision 0, approximately 1% of the cables in the plant are thermoplastic. These cables are primarily COAX used to support instrumentation. The subject ER is limited in scope and does not identify Fire PRA cables they may be thermoplastic and require a lower damage temperature. Significance of effects of different types of cables could be enough to impact results. Determine which (if any) fire PRA credited cables are thermoplastic and use appropriate damage threshold in fire damage scenarios.	CALC-ANOC-FP-09-00019, Rev. 0, EC-6964, “Safe Shutdown Cable Jacket Insulation Types at ANO,” evaluates the cables at both units and concludes that of the over 4600 cables reviewed, less that 0.3% have thermoplastic insulation. This calculation also confirms that thermoplastic cables are not used in power supply circuits. Thus, thermoplastic insulation at ANO is of minimal significance. CALC-ANOC-FP-09-0019 is referenced in the FPRA Scenarios Report.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	FSS-C8	Fire PRA Peer Review Finding FSS-C8-01 ANO has not confirmed that the wrap will not be subjected to direct flame impingement or validation that the wrap is qualified for fire impingement was not performed. Lack of basis for wrap validation is significant information; therefore, F&O is finding rather than suggestion. Review credited raceway wraps and either validate that wrap is qualified for fire impingement or validate that wrap is not subjected to fire impingement.	The FPRA analysis was revised such that it does not take credit for the 1-hour (Hemyc) fire wrap on service water pump cables in areas OO and B-6 or on charging pump cables in area DD. As shown in the Scenarios Report, in base scenarios for these areas, the wrapped cables are assumed to fail. For specific fire scenarios in these areas, the wrapped cables are assumed to fail if they are within the zone of influence of the fixed or transient ignition source. Thus, the FPRA results for these areas were calculated assuming that the 1-hour fire wrap does nothing to mitigate the extent of fire damage.	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.
A2-3736	Resolved	FSS-D3	Fire PRA Peer Review Finding FSS-D3-01 Implementation of the generic fire modeling methods did not consider the effect of fire growth and propagation within cable trays. A review of fire modeling applications within fire compartment 2109-U shows that the HGL determination was based upon bounding estimates for cabinet heat release rates and did not validate that the value would be bounding with the inclusion of fire spread to the cable trays. This is a finding because the analysis may not be bounding if the heat release rate contribution of the cable trays would be involved in a cabinet fire scenario. Consider the effect of fire growth and propagation within cable trays to validate that the value would be bounding with the inclusion of fire spread to the cable trays.	The generic fire modeling methods were updated to perform a case to account for hot gas layer effects on cable trays located one foot above the electrical panel. See Section 2.2 of the Multi-Compartment Analysis (PRA-ES-05-004). Therefore, the fire growth and propagation within the cable trays are addressed in the fire scenarios associated with bounding cabinet heat release rates such as with fire compartment 2109-U.	The Heat Release Rates used have been revised to incorporate data from NUREG-2178, Rev. 0, “Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLEFIRE) Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume, April, 2016. The use of this more recent information does not affect the original disposition to this F&O.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	FSS-D7	Fire PRA Peer Review Finding FSS-D7-01 Site-specific suppression system unavailability values were not evaluated. Surveillance requirements are incorporated into site procedures, but specific unavailability values in the cable spreading room are not tracked for Fire PRA purposes. The SR is not met per NUREG/CR-6850 guidance because the methodology does not include maintenance contributions to unavailability, credit for manual actuation of the system, dependent failures, and plant specific data. F&O is a finding since the analysis did not include attributes necessary to meet the NUREG/CR-6850 guidance for uncertainty evaluations. Incorporate the required plant specific data regarding unavailability per NUREG/CR-6850 guidance.	The ANO-2 Fire Scenarios Report (PRA-A2-05-003) discusses the review of the fire suppression systems credited in the Fire PRA. Explicit credit for suppression and detection systems is taken for the cable spreading room fire scenario only. The smoke detection system is credited for early detection of a fire in the cable spreading room. A review of impairments associated with the cable spreading room detection system indicated that only individual detectors were out of service for a limited period of time during the period from 2007 through 2009. Therefore, the unavailability of these systems is very low and is considered to be enveloped by the system unreliability data taken from NUREG/CR-6850.	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.
A2-3736	Resolved	FSS-D10 Other Affected SR FSS-H10	Fire PRA Peer Review Finding FSS-D10-01 Compartment 2111-T was one of those areas that was not walked down by ERIN, but was walked down by the FPRA Peer Review. A battery charger 2D33 is located in compartment 2111-T and is not listed in the Walk-down sheet or in the Ignition source data. Further review of the fire scenarios listed in ERIN Report 0247-06-0006.05 showed a fire scenario that takes into account battery charger 2D33 as the ignition source (FRANC scenario 2111-T-B in Attachment A).	The ignition source data sheet and walkdown sheet for compartment 2111-T were revised in CALC-08-E-0016-01 to include the battery charger. The walkdown sheets in CALC-08-E-0016-01 were developed during initial ignition frequency walkdowns. Subsequently, additional walkdowns were performed for fire scenario development. During the later walkdowns, access was obtained to some zones previously inaccessible.	The ignition frequencies have been revised to encompass NUREG 2169 Rev. 0, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009, January 2015.”	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 to Reflect New Guidelines	Importance to Application
A2-3736 *continued*	Resolved	FSS-D10 Other Affected SR FSS-H10 *continued*	It is recommended that the fire areas that were not walked down (listed in Table 2-1 of CALC-08-E-0016-01) have the ignition sources re-verified by a walkdown or compared against the list of fire scenarios. Because a fire scenario for the battery charger 2D33 was evaluated using the wrong ignition source frequency (see F&O FSS-A5-01), the battery charger needs to be added to the plant walk-down and the data sheets to ensure the correct ignition frequency is used in the CDF quantification. Because the missing data was due to not performing the plant walk-down, it is recommended that the list of compartments in Table 2-1 of CALC-08-E-0016-01 be re-considered for walk-downs to ensure that all the applicable ignition sources are adequately evaluated as fire scenarios.	The notes from the subsequent walkdowns were reviewed to determine if components in addition to those in CALC-08-E-0016-01 were noted. A review of the walkdown sheets resulted in some additional components being added based upon the walkdowns. In addition, other changes were identified and incorporated addressing new information relating to the components identified.	The use of this more recent information does not affect the original disposition to this F&O.	
A2-3736	Resolved	FSS-E2 Other Affected SR UNC-A2	Fire PRA Peer Review Finding FSS-E2-01 Section 7.1.2.1 of the ANO Fire Scenarios Report 0247-06-0006.05 evaluates the severity factor for ventilated cabinets located in fire zones equipped with automatic detection. The evaluation assumes that no target damage will result if the fire is suppressed within 30 minutes. This is an application of expert judgment in lieu of plant-specific data or generic estimates. The justification of the basis for this application has not been adequately developed or supported. Justification has not been provided for the basis of the specific expert judgment discussed above and therefore may not be considered to be valid.	The non-suppression probabilities for electrical cabinet fires were changed based upon a methodology that has been submitted to the EPRI Fire PRA Methods Panel. These values are based on Panel voltage ratings and do not include any assumptions for suppression times. Also see Entergy response to NRC PRA RAI 01, dated November 7, 2013 (2CAN111301) (ML13312A877).	Non-Suppression probabilities have been updated using NUREG 2169, “Rev. 0, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009, January 2015.”	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 to Reflect New Guidelines	Importance to Application
A2-3736 *continued*	Resolved	FSS-E2 Other Affected SR UNC-A2 *Continued*	Perform a more extensive study to justify the use of the 30-minute non-damage time based on fire brigade response.		Additionally, the treatment of fire propagation from electrical cabinets utilized FAQ 14-0009, “Treatment of Well Sealed MCC Electrical Panels Greater than 440V” for determining fire spread outside the cabinet. The use of this information addresses the finding and does not impact the disposition to this F&O.	
A2-3736	Resolved	FSS-G2	Fire PRA Peer Review Finding FSS-G2-01 Multi-Compartment Screening Analysis methodology includes a manual suppression probability based upon the rating of the fire barriers and that they would withstand a fire for a minimum of 90 minutes. A barrier failure probability is used based upon the bounding assumption that each barrier includes a normally closed door (7.4E-3). In the screening method, the value of manual suppression is multiplied by a barrier random failure probability. Considering that the manual suppression probability is based upon 90 minutes, these two values should not be ANDed. Given that the method is intended to be bounding for purposes of screening, this method may artificially reduce the CDF by up to 3 orders of magnitude.	This F&O was written on the analysis when credit for manual non-suppression probability was based on the rating of the barrier. This approach was corrected in the latest revision of the MCA/HGL analysis. The current analysis credits a manual non-suppression probability based on the time required to generate a hot gas layer. The original approach would be susceptible to a dependency between the non-suppression probability and the barrier failure probability. In the current approach the non-suppression probability is not related to the barrier rating and therefore has no significant dependency with respect to the barrier failure probability.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	FSS- H7 Other Affected SR FSS-E2	Fire PRA Peer Review Finding FSS- H7-01 Section 7.1.2.1 - Severity Factor for Ventilated / Open Cabinets from the Fire Scenarios Report describes values credited in fire brigade suppression prior to damage external to the effected cabinet. The assumptions related to this method have not been justified. The method assumes that fire detection will occur 30 minutes prior to external damage to cabinets. This assumption has not been justified. NUREG/CR-6850 indicates a bounding assumption that in-cabinet detectors may provide an additional 5 minutes of time for detection. Therefore, without justification 30 minutes should not be used. In addition, the 30 minutes does not account for plant specific response times to the alarm. This value would also need to be considered. The lower branch does provide a 15 minute delay to fire brigade response. The 15 minutes included in NUREG/CR-6850 is loosely based upon control room indication of fire due to failed equipment that results from a developed fire. Justification for credit of this value would need to be provided on how operations would know to send fire brigade based upon a fire limited to a single cabinet with no fire alarm.	The concern identified in this F&O no longer applies. The non-suppression probabilities for electrical cabinet fires were changed based upon a methodology that has been submitted to the EPRI Fire PRA Methods Panel. These values are based on Panel voltage ratings and do not include any assumptions for suppression times. Brigade response is only credited for transient fires in the cable spreading room. This is described in Section 9.0 of the Fire Scenarios Report (PRA-A2-05-003).	Non-Suppression probabilities have been updated using NUREG 2169, "Rev. 0, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009, January 2015.". Additionally, the treatment of fire propagation from electrical cabinets utilized FAQ 14-009, "Treatment of Well Sealed MCC Electrical Panels Greater than 440V" for determining fire spread outside the cabinet. The use of this information addresses the finding and does not impact the disposition to this F&O. The Brigade disposition response discussion has not changed and is still applicable.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	FSS- H7	Fire PRA Peer Review Finding FSS- H7-02 Section 9.1 of the Fire Scenarios Report contains the method used to evaluate overall compartment damage through hot gas layer development. This section of the report includes credit for fire brigade capability to alter hot gas layer development in 20 minutes which in essence results in a 1.0 probability that if a HGL (625 deg F) does not develop within 20 minutes it will not develop. This is an input/ assumption that is not clearly identified in the report. In addition, the basis for this value is not provided. The model employed does not discuss impacts that would support this time including fire brigade response time, fire characterization, and probabilities for failure.	Brigade response is only credited for transient fires in the cable spreading room. This is described in Section 9.0 of the Fire Scenarios Report (PRA-A2-05-003). This evaluation assumes that manual suppression would have to occur within 5 minutes and before damage to any cable trays. In this case, a hot gas layer would not have time to develop.	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.
A2-3736	Resolved	HRA-A2 Other Affected SR HRA-B3	Fire PRA Peer Review Finding HRA-A2-01 Identification of new operator actions for each fire scenario did not include fire area specific EOPs (Emergency Operating Procedures). This can result in the failure to identify operator actions which may de-energize equipment, lead to actions different from those assumed in the Fire PRA, and/or more direct symptom-based procedural actions. For example, for a fire in Fire Area B-3, the operator is specifically directed (by FIRES IN AREAS AFFECTING SAFE SHUTDOWN, 2203.049, Rev. 7) to verify the RCP breakers are OPEN with DC Control Power removed.	The new HRA events identified for the Fire PRA are documented in calculation PRA-A2- 05-002. This calculation evaluates the probabilities for these new operator actions and any new combinations between these actions and the other HRA events. Each of the HFEs that were assessed for the transition to NFPA-805 and contain a reference to the guiding procedures within the HRA worksheets. Revisions to OP-2203.049 will be required prior to transition to NFPA-805 that will provide additional guidance to the operators for fire in each area of the plant.	Update on OP-2203.049. This procedure has been updated and utilized in the development of the HRAs necessary to respond to a fire and calculate the HEPs associated with these HRAs.	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 to Reflect New Guidelines	Importance to Application
A2-3736 *continued*	Resolved	HRA-A2 Other Affected SR HRA-B3 *continued*	This is different from the HRA write-up which assumes that 10 minutes will be used to attempt to trip the RCPs in the CR and that "when they [the operators] tried to trip the RCPs from the control room, they would not be successful and would decide to open the associated breakers." This would be evaluated differently if procedurally guided. It was noted by the Fire PRA staff that the fire-related procedures were planned to be changed to make them more symptom-based and remove many of the steps currently in the procedures. The SR specifically requires, for each fire scenario, identification of any new fire-specific safe shutdown actions called out in the plant fire response procedures. The Fire PRA staff noted that the fire-related procedures were expected to be changed and did not want to add actions to the PRA that would have to be removed before applying the fire PRA. SR HRA-A2 was assessed as “Not Met” for the reasons discussed above. To “Meet” this SR, ANO-2 should update the fire-related procedures as the fire PRA staff indicated to be planned. Following this update to the fire-related procedures, perform and document a systematic review of the plant fire response procedures and identify fire-specific safe shutdown actions that may be taken by the operator for inclusion in the PRA. It is recommended that all HFEs included in the model be tied directly to either the fire-related procedures or the normal EOPs.	The transition to NFPA-805 will result in removal of many operator actions currently considered necessary under Appendix R. Therefore, many actions deemed necessary in the current version of OP-2203.049 are not required post transition to 805. Also see Entergy response to NRC PRA RAI 01, dated November 7, 2013 (2CAN111301) (ML13312A877).	Additionally, the HRA methodology used in determining the probability of failure for significant operator actions in the Fire PRA was revised to follow the guidance of NUREG 1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines. The use of this information does not impact the disposition to this F&O.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	HRA-A4	Fire PRA Peer Review Finding HRA-A4-02 Attachment C of the Scenarios Report (0247-06-0006.05 Revision 0) provides the results of talk- and walk-throughs with operations personnel regarding instrumentation and indication issues during fire scenarios. However, there is no evidence of reviews or talk-throughs with operations or training personnel to confirm that the interpretation is consistent with plant operational and training practices during fire scenarios. SR HRA-A4 requires at least a review of the interpretations of procedures associated with actions in the PRA with operations or training personnel to meet Capability Category I. SR HRA-A4 was assessed as Capability Category I for the reasons discussed above. To achieve Capability Category II/III, ANO-2 needs to perform and document the review or talk-through with operations or training personnel to confirm the interpretation of procedures is consistent with both operational and training practices. This should be performed after the fire-related procedures are revised. (See F&O HRA-A2-01)	Although this SR only meets CC-I, the issue identified does not impact the FPRA results and is ultimately considered a documentation issue associated with ensuring that the information provided by the Shift Managers involved with the interviews are familiar with the Ops and training requirements and expectations. ANO has a high degree of confidence that the Shift Managers are familiar with these expectations since it is one of the responsibilities of the Shift Managers. In addition to the general talk-throughs of fire scenarios and simulator observations in Attachment C of the Fire HRA calculation (PRA-A2-05-007), ANO-2 has documented operator interviews of operator actions identified for the Fire PRA in calculation PRA-A2-05-002 and has identified operator cues and instruments required for each operator action.	The HRA methodology used in determining the probability of failure for significant operator actions in the Fire PRA was revised to follow the guidance of NUREG 1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines. The use of this information addresses the finding and does not impact the disposition to this F&O.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	HRA-D2 Other Affected SR PRM-B9	Fire PRA Peer Review Finding HRA-D2-01 The ECCS vent path is modeled as a spurious open vent path (2CV-4740-2 and 2CV-4698-1) equivalent to a medium LOCA. The model logic (Gate FIRE003) includes an operator recovery action (RHFPZRVNTISO) from the control room to isolate the vent path by closing valve 2CV-4740-2. However, 2CV-4740-2 is one of the spuriously opened valves, and may not be remotely operable. The ECCS vent path may not be remotely isolable given a spurious opening due to a fire. The recovery action assigned in the model may not be effective. A detailed circuit analysis of 2CV-4740-2 should be performed.	Operator recovery action RHFPZRVNTISO, to close valve 2CV-4740-2, has been deleted from the FPRA model. Additionally, all valves requiring manual operation post fire have been evaluated for 92-18 concerns. If a valve has been identified to require manual manipulation after spurious operation and is indeed a 92-18 concern, the valve has been identified for modification within the FREs.	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.
A2-3736	Resolved	IGN-A5	Fire PRA Peer Review Finding IGN-A5-01 Table 3-2: some IEFs are not corrected to a "per reactor-year" basis. Some ignition frequencies are based on all modes of operation (e.g., batteries) and are correctly updated with calendar years as described in Section 3.1 of Entergy Calculation CALC-08-E-0016-01 Revision 0. However for those ignition frequencies, plant availability must be factored into the initiating event frequency or it will be conservative by approximately 10%. 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, 12-19, 23, 26, and 30. Multiply the frequencies of all-mode Ignition Frequencies by the availability factor (critical years/calendar years).	IGN-A5 is “Met” because the plant-specific fire frequency updates were revised to reflect a reactor-year basis. The plant availability was used in determining the frequencies by the fraction of time the plant was at-power. CALC-08-E-0016-01 Table 3-2 has been changed to show that all bins were updated on a reactor-year basis.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	IGN-B4 Other Affected SR IGN-A6	Fire PRA Peer Review Finding IGN-B4-01 It appears that the 5th and 95th frequencies in Table 3-2 of Entergy Calculation CALC-08-E-0016-01 Revision 0, does not match up with the 5th and 95th frequencies from Table C-3 of NUREG/CR-6850. Additionally, Bin 16 has been subdivided to include isophase bus ducts bins (16c and 16d) based on FAQ 0035, but this is not reflected in Table 3-2. Table 3-2 of Entergy Calculation CALC-08-E-0016-01 Revision 0 does not accurately reflect the data used to develop ignition frequencies and compartment initiating event frequencies. Additionally, a spot check of the uncertainty information associated with the generic frequencies provided by the PRA team (FIF Bart Input Template (Updated 11-14-2008).xls) found one error (Bin 1 Range Factor is listed as 11.0 and it should be 10.0); other data appears to be correct. Correct Table 3-2 to reflect the correct information used for the Bayesian Update. Review all data used for the Bayesian update and correct any additional errors (beyond Bin 1), if found. It is also recommended that the document specify the prior distribution parameters used in the BART calculations (i.e., mean and range factor) to allow for reproduction of the update results. Provide details on isophase bus duct frequencies (bins 16c and 16d) from FAQ 0035.	Table 3-2 of CALC-08-E-0016-01 was revised to reflect the data from NUREG/CR-6850 Table C-3, as well as data from FAQ 07-0035 for bins 16c and 16d and data from FAQ 06-0017 for bins 16a and 16b. Per e-mail from the peer reviewer for this SR (L. Shanley) to J. Renner, dated 9/2/09, 1:36 pm; the portion of this F&O related to the uncertainty information was in error. No action is required.	The ignition frequencies have been revised to encompass NUREG 2169 Rev. 0, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009, January 2015.” The use of this information addresses the finding and does not impact the disposition to this F&O.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	PRM-B2	Fire PRA Peer Review Finding PRM-B2-01 The majority of the Internal Event deficiencies are open. The items have not been dispositioned such that their impact on the Fire PRA could be determined. A review of the items indicate one item in particular (timing for securing RCPs following a loss of CCW) that could have impact on the Fire PRA results. ANO-2 needs to review the disposition of open items from their level 1 internal initiator PRA to ensure that their disposition remains valid in view of the unique aspects of fires.	The ANO-2 Internal Events Peer Review (LTR-RAM-II-08-020 identified 26 findings against the Internal Events supporting requirements. These findings are discussed in Attachment U of this LAR submittal. A qualitative review indicated that most of these findings would have no or minor impact on fire risk. ANO-2 has created a new Fire HFE associated with tripping the RCP outside the control room. The system time window for this HFE uses a shorter available time than that assumed in the internal events analysis. This new available time is based upon WCAP16175-P-A "Model for Failure of RCP Seals Given Loss of Seal Cooling in CE NSSS Plants." The system time window for the HFE associated with tripping the RCP from within the control room (developed as part of the internal events PRA) will be revised in the next revision to the internal events PRA. The difference in failure probability using the shorter available time for this HFE does not change significantly due to the contribution of the Caused Based Method to the resulting failure probability.	The HRA methodology used in determining the probability of failure for significant operator actions in the Fire PRA was revised to follow the guidance of NUREG 1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines. The use of this information does not impact the disposition to this F&O.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	PRM-B9 Other Affected SR HRA-D2	Fire PRA Peer Review Finding PRM-B9-01 Components credited to support fire recovery actions have not been reviewed per the requirements of the applicable internal events standard. For example, valves added to support recovery actions that isolate spurious operation flow diversions do not always included credited power supply dependencies and random failure modes for the equipment. Review added components credited in HRA recovery and add appropriate basic events and links within the PRM. Document as required.	All new fire recovery actions in the ANO-2 FPRA model have been reviewed per the requirements of the applicable internal events standard and appropriate basic events and links were added to the plant response model to ensure that dependencies are modeled. Therefore, plant response model (PRM)-B9 is “Met.”	The HRA methodology used in determining the probability of failure for significant operator actions in the Fire PRA was revised to follow the guidance of NUREG 1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines. The use of this information does not impact the disposition to this F&O.	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	PRM-C1	Fire PRA Peer Review Finding PRM-C1-01 The Supporting Requirement requires the Fire plant response model be documented consistent with HLR-IE-D, HLR-AS-C, HLR-SC-C, HLR-SY-C, and HLR-DA-E and their SRs in Section 2. No specific documentation could be found that explains the development of the Plant Response Model consistent with the SR requirements above. The supporting requirement to document the PRM (PRM-C1) is not met. The work supporting the PRM (including modifications to the internal events PRA) needs to be tied together in a summary document. Prepare a calculation report that documents the PRM in accordance with SR PRM-C1.	This is a documentation issue only. The following discussion was added to Section 1 of the Component and Cable Selection reports linking the requirements of PRM-C1 to the internal events model. "The FPRA model uses the ANO Internal Events model as its basis. The methodology used to update the ANO Internal Events model to generate the FPRA model uses the same methodology used in the development and documentation of the Internal Events model. The FPRA documentation may differ from that used for the internal events model due to FPRA specific requirements, however, the basic intent of the internal events documentation requirements are met. The FPRA therefore meets the same model development and documentation requirements applicable to the internal events model. Therefore, requirements of the ANSI/ASME standard applicable to the PRA model specified in PRM-C1 are met via the internal event model meeting the associated model requirements." Additional details on development of the PRM from the Internal Events PRA model can be found in Section 4 of the ANO-2 Component and Cable Selection Report. Also see Entergy response to NRC PRA RAI 01, dated November 7, 2013 (2CAN111301) (ML13312A877).	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 to Reflect New Guidelines	Importance to Application
A2-3736	Resolved	UNC-A1 Other Affected SR FQ-F1, UNC-A2	Fire PRA Peer Review Finding UNC-A1-01 The impact of parameter uncertainties were not estimated or propagated as required in SR QU-E1 and for LERF (LE-F3). Additionally, the uncertainties are not characterized as required by SR QU-E4. Appendix D lists and discusses sources of uncertainty; however the characterization of these uncertainties is not detailed. Uncertainty must be evaluated to include "ESTIMATE the uncertainty interval of the CDF results" per QU-E1, Capability Category I or II (different details, but both require uncertainty interval to be estimated). This is also required for LERF (LE-F3). Propagate or estimate the impact of parameter uncertainties on CDF and LERF. Alternatively, a defined basis can be developed to support the non-applicability of this SR.	Appendix D of the Fire PRA Summary Report (PRA-A2-05-004) addresses sensitivity to the sources of uncertainty for the Fire PRA tasks. Additionally, PRA-A2-05-006, “ANO-2 Fire PRA Uncertainty/Sensitivity Analysis” has been developed to calculate the uncertainty associated with the CDF and LERF values of the Fire PRA. The information provided in these two documents satisfies the requirements identified in this F&O. Also see Entergy response to NRC PRA RAI 01, dated November 7, 2013 (2CAN111301) (ML13312A877).	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.
A2-3736	Resolved	MU-B4	Fire PRA Peer Review Finding MU-B4-01 There is no reference to a peer review for upgrades. Did not find a section which addressed upgrades (not updates) to the PRA specifically involving changes to key PRA software.	EN-DC-151 (PSA Maintenance and Update) procedure was revised to require an industry peer review if a PSA upgrade per ASME/ANS RA-Sa-2009 has occurred.	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 to Reflect New Guidelines	Importance to Application
N/A	Resolved	FSS-D8	2011 Focused-Scope Peer Review Finding FSS-D8-01 This SR requires an assessment of fire detection and suppression system effectiveness in the context of analyzed fire scenarios. The ANO-2 fire PRA credits the smoke-actuated dry pipe sprinkler system in the cable spreading room for preventing target damage from a transient fire. That is, the fire PRA only models target damage if the system fails. ANO-2 did not perform (or at least did not document) a technical basis that the suppression system is capable of preventing target damage. With the current model assuming a 317 kW fire, a relatively short fire growth phase, and the relative proximity of overhead cable trays to the postulated fire, the suppression system may not be capable of suppression prior to target damage. • Suggested Resolution – Provide a technical basis that the cable spreading room suppression system can effectively extinguish a transient fire prior to target damage. As an alternate resolution, consider making this area a “zero-transient” area per EN-DC-161 such that a 69 kW transient (versus the current 317 kW) can be modeled per the ANO-2 transient methodology, in addition to being able to credit prompt suppression if the procedure revision requires a continuous fire watch when transient combustibles are present. The technical basis that the suppression system can extinguish the 69 kW fire prior to target damage should also be provided per this Supporting Requirement.	The lower heat release rate has been justified as applicable in the cable spreading rooms and will be supported by a revision to EN-DC-161, which will restrict the combustibles allowed within the affected rooms. A Work tracking item (LO-WTANO- 2010-00222 CA 00011) has been issued to ensure that this zone is included as a no transient zone. This action requires an update to the procedure for control of combustibles [EN-DC-161] to ensure "zerotransient" zones in the ANO-1 and ANO-2 Fire Scenarios Reports are designated and maintained. As part of the transition to an NFPA-805 licensing basis, the ANO-1 and ANO-2 Fire Scenarios Reports specify that certain fire zones must be maintained as "zero-transient" zones. Credit for manual suppression of a 69 kW transient fire has been credited for this fire zone. No specific credit for the automatic suppression systems has been taken. Also see Entergy response to NRC PRA RAI 01, dated November 7, 2013 (2CAN111301) (ML13312A877).	LO-WTANO-2010-00222 CA-00011 has been completed and EN-DC-161 has been revised to designate zero combustible areas for both units as part of transition to NFPA-805. The update relating to the Work Tracking item provides the current status of that item, all else in the 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 to Reflect New Guidelines	Importance to Application
N/A	Resolved	FSS-H1	2011 Focused-Scope Peer Review Finding FSS-H1-01 The ANO-2 fire PRA documentation is generally adequate and this SR is met with an F&O. Procedure EN-DC-161 “Control of Combustibles” Rev 5 was reviewed against Table 8-1 of the Fire Scenarios Report. This review identified that four fire zones (2111-T, 2096-M, 2112-BB, and 2182-J) allowed transient combustibles (in some cases up to 300 pounds) with a roving fire watch. However, the fire PRA model postulates 69 kW transient fires in these areas consistent with its methodology for “zero-transient” areas. After some discussion, it was identified that ANO-2 plans to revise EN-DC-161 to require continuous fire watch for any transient storage in these areas. The Fire Scenarios Report should be revised to reference an ANO-2 action item or CR tracking implementation of this procedure revision. On a related note, ANO-2 may consider crediting prompt suppression for these areas. Suggested Resolution – Revise the Fire Scenarios Report to reference an ANO-2 action item or CR tracking implementation of revision to ENDC-161.	A corrective action has been written to revise EN-DC-161 to address this discrepancy. Work tracking item #LO-WTANO-2010-00222 CA-00011 has been issued to ensure that this zone is included as a no transient zone. This action requires an update to the procedure for control of combustibles [EN-DC-161] to ensure "zero-transient" zones in the ANO-1 and ANO-2 Fire Scenarios Reports are designated and maintained. As part of the transition to an NFPA-805 licensing basis, the ANO-1 and ANO-2 Fire Scenarios Reports specify that certain fire zones must be maintained as "zero-transient" zones. While the action item has been written and incorporated into the Paperless Condition Reporting System (PCRS), it was not deemed prudent to incorporate a temporary work tracking item into the Fire Scenario Report.	LO-WTANO-2010-00222 CA-00011 has been completed and EN-DC-161 has been revised to designate zero combustible areas for both units as part of transition to NFPA-805. The update relating to the Work Tracking item provides the current status of that item, all else in the 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 to Reflect New Guidelines	Importance to Application
N/A	Resolved	IGN-A7	2011 Focused-Scope Peer Review Finding IGN-A7-01 This SR relates to the fire ignition frequency apportioning methodology. ANO-2 meets this SR for transient ignition sources based on preponderance of evidence; however, one deficiency was noted with the turbine building transient fire frequency. In the turbine building, ANO-2 postulated 12 transient fire scenarios that could affect PRA targets. The area factor for each source was calculated as 100 sq. ft. divided by the total turbine building floor area. The total area factor for all turbine building transients was therefore 1,200 sq. ft. divided by the total turbine building floor area. Based on inspection of plant drawings, the total floor area containing PRA targets of concern is greater than 1,200 sq. ft., and the ANO-2 model is therefore not accounting for some fraction of the turbine building transient fire frequency. The underlying cause of this error was using a nominal 100 sq. ft. for each transient and not postulating enough scenarios to cover the risk-relevant floor area. Alternatively, the area factor for each transient could be inflated to ensure the entire risk-relevant floor area is considered. • Suggested Resolution – Adjust the turbine building transient area factors such that the total floor area containing risk-relevant targets is considered.	The floor areas assumed for the transient fires were adjusted based upon walkdowns and drawings to better reflect the area of the scenarios being developed. This approach increased the size of the floor area used in calculating the ignition frequency for each of the scenarios considered. This new information is documented in the Scenario Report Attachment H and Attachment D. Also see Entergy response to NRC PRA RAI 01, dated November 7, 2013 (2CAN111301) (ML13312A877).	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 to Reflect New Guidelines	Importance to Application
A2-5110	Resolved	HR-G3	2014 Focused-Scope Peer Review Finding HR-G3-01 All assumptions that impact feasibility of operator actions need to be validated before the HEPs can be applied, specifically 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. Related to the above are assumptions made with regards to explicit modeling of instrumentation. Assumptions regarding procedure changes also need to be validated.	Credit for instrumentation is documented in the post fire safe shutdown analysis. This instrumentation is correlated to the credited HEPs in the Detailed Fire HRA for Selected ANO-2 HFE, Appendices B and C.	The HRA methodology used in determining the probability of failure for significant operator actions in the Fire PRA was revised to follow the guidance of NUREG 1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines. The use of this information does not impact the disposition to this F&O. Although the specific sections depicted in the Disposition have changed due to revisions in the HRA analysis, the information provided is still appropriate.	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 to Reflect New Guidelines	Importance to Application
A2-5111	Resolved	HR-G7	2014 Focused-Scope Peer Review Finding HR-G7-01 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.	A review was performed of the most challenging scenario with respect to manpower requirements, control room abandonment, which confirmed that available manpower is sufficient to support multiple HEPs in a cutset for the control room abandonment cutsets. The review is documented in ANO-2 PRA-A2-05-007, Rev. 1, "Fire Probabilistic Risk Assessment Human Reliability Analysis (HRA) Notebook," Section 4.8.	The HRA methodology used in determining the probability of failure for significant operator actions in the Fire PRA was revised to follow the guidance of NUREG 1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines. The use of this information does not impact the disposition to this F&O.	This issue was resolved and therefore has no impact in STI change evaluations performed in accordance with the SFCP.

Table 3

List of SRs Assessed as CC-I in the ANO-2 Fire PRA Model

SR	Topic	Status	Changes to Modeling Elements to Reflect New Guidelines	Importance to Application
PP-B2 / PP-B3	Credit for non-rated fire barriers/ Credit for spatial separation as a partitioning feature	The ANO Plant Partitioning and Fire Ignition Frequency Development report uses the plant areas that are identified in the plant FPP. The ANO-2 Peer Review team noted two instances where this approach allows credit for non-rated fire barriers: 1) the glass partition between the ANO-1 and ANO-2 Control Rooms and 2) the Turbine deck, which does not have a rated barrier between the units. The MCR Abandonment analysis provides an evaluation of fires that would fail the partition and require both Control Rooms to be abandoned. The MCA evaluated fires on the turbine deck to determine if a fire in this area would spread to the other unit. The MCA did not identify any fires on the turbine deck that would spread to the other unit. Detailed evaluation of the impact to fires on the barriers was completed. The detailed evaluation of the non-rated barriers ensures the Category I limitation does not impact the results or conclusions of the model.	No change. Original Disposition remains valid.	This issue has no impact in STI change evaluations performed in accordance with the SFCP.
PP-B5	Credit for active fire barriers	The ANO Plant Partitioning was performed without taking credit for active fire barriers. Active barriers are used to separate divisional rooms. Not crediting the active barriers for partitioning only shifts the burden for fire risk evaluation to the Fire Scenario development process. Therefore, this assumption has a minor impact on overall fire risk. Active fire barriers were appropriately credited/modeled in scenario development. While this approach only meets the Category I requirement for portioning (PP), it does not significantly impact the overall model results or conclusions.	No change. 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
CS-B1	Analyze Electrical Buses for Overcurrent Coordination	Section 4.4 of the Component and Cable Selection Report (PRA-A2-05-005) documents the Electrical Coordination/Protection for ANO-2. This document refers to Upper Level Document ULD-0-TOP-12 for FPRA components in the SSEL. Components that are not in the SSEL are evaluated in Table 4-3 of the calculation. The difference between Category I and Category II is the use of an existing document (Cat I) instead of completing a new analysis for all modeled buses (Cat II). The Category I classification only partially applies. An existing analysis was used for FPRA components in the SSEL. Evaluation of non- SSEL buses meets the Category II requirement. Though the current method only partially meets the Category II requirement, it was judged to be acceptable for the NFPA 805 application. Also see Entergy response to NRC PRA RAI 01, dated November 7, 2013 (2CAN111301) (ML13312A877).	No change. Original Disposition remains valid.	This issue has no impact in STI change evaluations performed in accordance with the SFCP.
IGN-A10	Provide uncertainty intervals for fire ignition frequencies	Section 6.2 of the ANO-2 FPRA Uncertainty/Sensitivity Analysis (PRA-A2-05-006) discusses the uncertainty intervals used for the fire ignition frequencies for propagating uncertainty. The uncertainty from ignition frequency development is primarily from the NUREG guidance. ANO strictly followed NUREG/CR-6850 guidance in developing ignition frequencies using numbers included in the document.	The ignition frequencies have been revised to encompass NUREG-2169 Rev. 0, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009, January 2015.” The use of this information does not impact the disposition to this SR.	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-2 Individual Plant Examination of External Events (IPEEE) submittal in response to the NRC IPEEE Program (Reference 12). 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-2 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 may be used to assess external event hazard risk at ANO-2 for STI changes.

Because ANO-2 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-2 shutdown safety program developed to support implementation of NUMARC 91-06 (Reference 17) 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-2 shutdown safety program includes input from a Defense-in-Depth shutdown EOOS PRA model.

4. CONCLUSIONS

The information presented herein demonstrates that the ANO-2 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 1. In Entergy letter 2CAN061406 (Reference 18), the response to PRA RAI 20 explains that, after a detailed review was performed, the changes in the SRs between ASME RA-Sb-2005 and ASME/ANS RA-SA-2009, and changes between RG 1.200, Revision 1 and 2, do not invalidate the ANO-2 peer review or change any of the findings and observations.

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. WCAP-15910, “Arkansas Nuclear One, Unit 2: Probabilistic Risk Assessment Peer Review Report,” November 2002.

6. LTR-RAM-II-08-020, “RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for the Arkansas Nuclear One, Unit 2 Probabilistic Risk Assessment,” July 2008.

7. ENTGANO150-REPT-002, “Arkansas Nuclear One Unit 2 Internal Flooding Probabilistic Risk Assessment Peer Review,” Revision 0, April 2017.

8. PSA-ANO2-01-PPCR, “Plant and Procedure Change Review (PPCR),” Revision 0, November 2016.

9. LTR-RAM-II-09-046, “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 2 Fire Probabilistic Risk Assessment,” September 2009.

10. LTR-RAM-II-11-064, “Focused Scope Fire PRA Peer Review for Arkansas Nuclear One Unit 2,” December 2011.

11. “Arkansas Nuclear One – Unit 2 Fire HRA Peer Review,” Curtiss-Wright Scientech, June 2014.

12. Generic Letter 88-20, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f), Supplement 4,” June 1991.

13. 2CAN081401, “License Amendment Request to Adopt NFPA-805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition),” Arkansas Nuclear One – Unit 2, Docket No. 50-368, License No. NPF-6, December 2012 (ADAMS Accession No. ML14219A635).

14. 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.

15. PRA-A2-05-004, “ANO-2 Fire PRA Summary Report,” Revision 2, March 2015.

16. NUREG/CR-6850 – EPRI-1011089, “Fire PRA Methodology for Nuclear Power Facilities,” August 2005.

17. NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management,” December 1991.

18. 2CAN061406, “Response to Request for Additional Information Adoption of National Fire Protection Association Standard NFPA-805,” Arkansas Nuclear One, Unit 2, Docket No. 50-368, License No. NPF-6, June 2014.

19. “ANO-2 Focus Scope Peer Review ANO-2 Fire PRA FSS-A, C, D, E and H,“ Kazarians & Associates, Inc., November 2012.

20. “Arkansas Nuclear One, Unit 2 Fire PRA Focus Scope Peer Review Report,“ JENSEN HUGHES, September 2016.

ATTACHMENT 3 to

2CAN021802

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or controlled leakage.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary to secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

AZIMUTHAL POWER TILT – T_q

1.17 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

DOSE EQUIVALENT I-131

1.18 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same committed effective dose equivalent (CEDE) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The CEDE dose conversion factors used to determine the DOSE EQUIVALENT I-131 shall be performed using Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

DOSE EQUIVALENT XE-133

1.19 DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

~~STAGGERED TEST BASIS~~

1.20 Deleted~~A STAGGERED TEST BASIS shall consist of:~~

~~a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and~~

~~b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.~~

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

~~THIS PAGE INTENTIONALLY LEFT BLANK~~

TABLE 1.1

OPERATIONAL MODES

Moved from TS Page 1-8

MODE	REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 300 °F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 300 °F
3. HOT STANDBY	< 0.99	0	$\geq$ 300 °F
4. HOT SHUTDOWN	< 0.99	0	300 °F > T_{avg} > 200 °F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200 °F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140 °F

* Excluding decay heat.

** Reactor vessel head unbolted or removed and fuel in the vessel.

TABLE 1.2

FREQUENCY NOTATION

Moved from TS Page 1-9

NOTATION	FREQUENCY
~~S~~	~~At least once per 12 hours.~~
~~D~~	~~At least once per 24 hours.~~
~~W~~	~~At least once per 7 days.~~
~~M~~	~~At least once per 31 days.~~
~~Q~~	~~At least once per 92 days.~~
~~TA~~	~~At least once per 123 days.~~

~~SA~~	~~At least once per 184 days.~~
~~R~~	~~At least once per 18 months.~~
S/U	Prior to each reactor startup~~.~~
~~P~~	~~Completed prior to each release.~~
N.A.	Not applicable~~.~~
SFCP	In accordance with the Surveillance Frequency Control Program

Moved to TS Page 1-7

~~TABLE 1.1~~

~~OPERATIONAL MODES~~

~~MODE~~	~~REACTIVITY CONDITION, K_{eff}~~	~~%RATED THERMAL POWER*~~	~~AVERAGE COOLANT TEMPERATURE~~
~~1. POWER OPERATION~~	~~$\geq$ 0.99~~	~~> 5%~~	~~$\geq$ 300 °F~~
~~2. STARTUP~~	~~$\geq$ 0.99~~	~~$\leq$ 5%~~	~~$\geq$ 300 °F~~
~~3. HOT STANDBY~~	~~< 0.99~~	~~0~~	~~$\geq$ 300 °F~~
~~4. HOT SHUTDOWN~~	~~< 0.99~~	~~0~~	~~300 °F > T_{avg} > 200 °F~~
~~5. COLD SHUTDOWN~~	~~< 0.99~~	~~0~~	~~$\leq$ 200 °F~~
~~6. REFUELING**~~	~~$\leq$ 0.95~~	~~0~~	~~$\leq$ 140 °F~~

~~* Excluding decay heat.~~

~~** Reactor vessel head unbolted or removed and fuel in the vessel.~~

Moved to TS Page 1-7

~~TABLE 1.2~~

~~FREQUENCY NOTATION~~

~~NOTATION~~	~~FREQUENCY~~
~~S~~	~~At least once per 12 hours.~~
~~D~~	~~At least once per 24 hours.~~
~~W~~	~~At least once per 7 days.~~
~~M~~	~~At least once per 31 days.~~
~~Q~~	~~At least once per 92 days.~~
~~TA~~	~~At least once per 123 days.~~
~~SA~~	~~At least once per 184 days.~~
~~R~~	~~At least once per 18 months.~~
~~S/U~~	~~Prior to each reactor startup.~~
~~P~~	~~Completed prior to each release.~~
~~N.A.~~	~~Not applicable.~~

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN – T_{avg} > 200 °F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With the SHUTDOWN MARGIN less than that required above, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

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

b. When in MODES 1 or 2#, in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~ by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of (e) below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

---

* See Special Test Exception 3.10.1.

With $K_{eff} \geq 1.0$.

With $K_{eff} < 1.0$.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, in accordance with the Surveillance Frequency Control Program~~at least once per 24 hours~~ by consideration of at least the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% $\Delta k/k$ in accordance with the Surveillance Frequency Control Program~~at least once per 31 Effective Full Power Days (EFPD)~~. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than that required above, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

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

b. In accordance with the Surveillance Frequency Control Program~~At least once per 24 hours~~ by consideration of at least the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be ≥ 2000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the reactor coolant system < 2000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be ≥ 2000 gpm within one hour prior to the start of and in accordance with the Surveillance Frequency Control Program~~at least once per hour~~ during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump or containment spray pump is in operation as a shutdown cooling pump and supplying ≥ 2000 gpm through the reactor coolant system.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be ≥ 540 °F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 540 °F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be: ≥ 540 °F in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

With K_{eff} ≥ 1.0.

* See Special Test Exception 3.10.5.

ACTION: (Continued)

e. With more than one CEA misaligned from any other CEA in its group by more than 7 inches (indicated position), be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

4.1.3.1.2 Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction in accordance with the Surveillance Frequency Control Program~~at least once per 92 days~~. ~~(Note 1)~~

Note 1 - Movement of CEA 4 is not required for the remainder of Cycle 26. If an outage of sufficient duration occurs prior to the end of Cycle 26, maintenance activities will be performed to restore the CEA.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5 inches of each other in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each CEA not fully inserted.

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

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program~~at least once per 18 months~~.

---

* With the reactor trip breakers in the closed position.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.7 seconds and the arithmetic average of the CEA drop times of all CEAs, from a fully withdrawn position, shall be $\leq$ 3.2 seconds from when the electrical power is interrupted to the CEA drive mechanisms until the CEAs reach their 90 percent insertion positions with:

a. $T_{avg} \geq 525$ °F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the CEA drop times determined to exceed either of the above limits, restore the CEA drop times to within the above limits prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of all CEAs shall be demonstrated through measurement prior to reactor criticality:

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

b. For specifically affected individuals CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to the Full Out position.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than the Full Out position, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Withdraw the CEA to the Full Out position, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to the Full Out position:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. In accordance with the Surveillance Frequency Control Program~~At least once per 12 hours thereafter~~.

---

* See Special Test Exception 3.10.2.

With $K_{eff} \geq 1.0$.

ACTION: (Continued)

b) Both CEACs inoperable:

Be in at least HOT STANDBY within 8 hours of exceeding the Transient Insertion Limit.

b. With the regulating CEA groups or Group P CEAs inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, either:

1. Restore the regulating groups or Group P CEAs to within the Long Term Steady State Insertion Limit within two hours, or

2. Be in at least HOT STANDBY within the next 6 hours.

c. With the regulating CEA groups or Group P CEAs inserted between the Short Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 4 hours per 24 hour interval, operation may proceed provided any subsequent increase in thermal power is restricted to ≤ 5% of rated thermal power per hour.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group and Group P CEAs shall be determined to be within the Transient Insertion Limits in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~ except during time intervals when the PDIL Alarm is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups or Group P CEAs are inserted beyond the Long Term Steady State Insertion Limit or the Short Term Steady State Insertion Limit but within the Transient Insertion Limit shall be determined in accordance with the Surveillance Frequency Control Program~~at least once per 24 hours~~.

3/4. 2 POWER DISTRIBUTION LIMITS

3/4.2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate limit shall be maintained by either:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

b. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC Channel (when COLSS is out of service).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

a. With COLSS in service and the linear heat rate limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on linear heat rate, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limit and either:

   1. Restore the linear heat rate to within its limits within 1 hour of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

b. With COLSS out of service and the linear heat rate limit not being maintained as indicated by operation outside the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT, either:

   1. Restore the linear heat rate to within its limits within 2 hours of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying in accordance with the Surveillance Frequency Control Program~~at least once per 2 hours~~ that the linear heat rate, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.

4.2.1.3 In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on linear heat rate.

POWER DISTRIBUTION LIMITS

RADIAL PEAKING FACTORS

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*

ACTION:

With a F^m_{xy} exceeding a corresponding F^c_{xy}, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to PLANAR RADIAL PEAKING FACTOR by a factor equivalent to $\geq F^m_{xy}/F^c_{xy}$ and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0] \times 100\%$ is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}); or

c. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}), obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. In accordance with the Surveillance Frequency Control Program~~At least once per 31 days of accumulated operation~~ in MODE 1.

* See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS

4.2.3 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~ when the COLSS is inoperable.

c. Verifying in accordance with the Surveillance Frequency Control Program~~at least once per 31 days~~, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors in accordance with the Surveillance Frequency Control Program~~at least once per 31 days~~ to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying in accordance with the Surveillance Frequency Control Program~~at least once per 2 hours~~ that the DNBR, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.

4.2.4.3 In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

POWER DISTRIBUTION LIMITS

RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 120.4 x 10^6 lbm/hr.

APPLICABILITY: MODE 1

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The Reactor Coolant Cold Leg Temperature (T_c) shall be maintained between 542 °F and 554.7 °F.

APPLICABILITY: MODE 1 above 30% of RATED THERMAL POWER.

ACTION:

With the Reactor Coolant Cold Leg Temperature exceeding its limit, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The Reactor Coolant Cold Leg Temperature shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limits in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~ using the COLSS or any OPERABLE Core Protection Calculator channel.

* See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The average pressurizer pressure shall be maintained between 2025 psia and 2275 psia.

APPLICABILITY: MODE 1.

ACTION:

With the average pressurizer pressure exceeding its limits, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The average pressurizer pressure shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program~~at least once per 18 months~~ during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program~~at least once per 18 months~~. Neutron detectors are exempt from response time testing.~~ Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.~~

4.3.1.1.4 The Core Protection Calculator System shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~ by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

4.3.1.1.5 The affected Core Protection Calculator Channel shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a valid CPC Cabinet High Temperature alarm.

TABLE 4.3-1

REACTOR PROTECTION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U (1)	N.A.
2. Linear Power Level – High	SFCP	SFCP~~D~~ (2,3,4) ~~M (3,4)~~ ~~Q (4)~~	SFCP~~TA (10)~~	1,2
3. Logarithmic Power Level – High	SFCP	SFCP~~R~~ (4)	SFCP~~TA (10)~~ S/U (1)	1,2,3*,4*,5*
4. Pressurizer Pressure – High	SFCP	SFCP~~R~~	SFCP~~TA (10)~~	1,2
5. Pressurizer Pressure – Low	SFCP	SFCP~~R~~	SFCP~~TA (10)~~	1,2
6. Containment Pressure – High	SFCP	SFCP~~R~~	SFCP~~TA (10)~~	1,2
7. Steam Generator Pressure – Low	SFCP	SFCP~~R~~	SFCP~~TA (10)~~	1,2
8. Steam Generator Level – Low	SFCP	SFCP~~R~~	SFCP~~TA (10)~~	1,2
9. Local Power Density – High	SFCP	SFCP~~D~~ (2,4~~)~~ ~~R (~~4,5)	SFCP~~TA (10)~~ ~~R~~ (6)	1,2
10. DNBR – Low	SFCP	SFCP~~S~~ (2,4,5,7)~~,~~ ~~D (2,4),~~ ~~R (4,5)~~	SFCP~~TA (10)~~ ~~R~~ (6)	1,2
11. Reactor Protection System Logic	N.A.	N.A.	SFCP~~TA (10)~~	1,2,3*,4*,5*
12. Reactor Trip Breakers	N.A.	N.A.	SFCP~~M~~	1,2,3*,4*,5*
13. Core Protection Calculators	SFCP	SFCP~~D~~ (2,4,5) ~~R (4,5)~~	SFCP~~TA~~ (6,9~~,10~~) ~~R (6)~~	1,2
14. CEA Calculators	SFCP	SFCP~~R~~	SFCP~~TA (10)~~ ~~R~~ (6)	1,2

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and, if necessary, adjust the CPC flow calibration addressable constant FC1 such that each CPC indicated flow is less than or equal to the measured flow rate.

(8) - Deleted

(9) - The CPC CHANNEL FUNCTIONAL TEST shall include the verification that the correct values of addressable constants are installed in each OPERABLE CPC.

~~(10) - On a STAGGERED TEST BASIS.~~

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program~~at least once per 18 months~~ during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program ~~at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3~~.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNELS CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP~~R~~	N.A.
b. Containment Pressure – High	SFCP	SFCP~~R~~	SFCP~~TA(2)~~	1,2,3
c. Pressurizer Pressure – Low	SFCP	SFCP~~R~~	SFCP~~TA(2)~~	1,2,3
d. Automatic Actuation Logic	N.A.	N.A.	SFCP ~~TA~~(1~~,2~~)	1,2,3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP~~R~~	N.A.
b. Containment Pressure -- High - High	SFCP	SFCP~~R~~	SFCP~~TA(2)~~	1,2,3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP ~~TA~~(1~~,2~~)	1,2,3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP~~R~~	N.A.
b. Containment Pressure -- High	SFCP	SFCP~~R~~	SFCP~~TA(2)~~	1,2,3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP ~~TA~~(1~~,2~~)	1,2,3
4. MAIN STEAM AND FEEDWATER ISOLATION (MSIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP~~R~~	N.A.
b. Steam Generator Pressure – Low	SFCP	SFCP~~R~~	SFCP~~TA(2)~~	1,2,3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP ~~TA~~(1~~,2~~)	1,2,3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNELS CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. CONTAINMENT COOLING (CCAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP~~R~~	N.A.
b. Containment Pressure – High	SFCP	SFCP~~R~~	SFCP~~TA(2)~~	1,2,3
c. Pressurizer Pressure – Low	SFCP	SFCP~~R~~	SFCP~~TA(2)~~	1,2,3
d. Automatic Actuation Logic	N.A.	N.A.	SFCP ~~TA~~(1~~,2~~)	1,2,3
6. RECIRCULATION (RAS)				
a. Manual (Trip Buttons) (a)	N.A.	N.A.	SFCP~~R~~	N.A.
b. Refueling Water Tank – Low	SFCP	SFCP~~R~~	SFCP~~TA(2)~~	1,2,3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP ~~TA~~(1~~,2~~)	1,2,3
7. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	SFCP	SFCP~~R~~	SFCP~~R~~	1,2,3
b. 460 volt Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP~~R~~	SFCP~~R~~	1,2,3
8. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Switches)	N.A.	N.A.	SFCP~~R~~	N.A.
b. SG Level and Pressure (A/B) – low and ΔP (A/B) – High	SFCP	SFCP~~R~~	SFCP~~TA(2)~~	1,2,3
c. SG Level (A/B) – Low and No Pressure – Low Trip (A/B)	SFCP	SFCP~~R~~	SFCP~~TA(2)~~	1,2,3
d. Automatic Actuation Logic	N.A.	N.A.	SFCP ~~TA~~(1~~,2~~)	1,2,3

Table 4.3-2 (Continued)

TABLE NOTATION

(a) Remote manual not provided for RAS. These are local manuals at each ESF auxiliary relay cabinet.

(1) The logic circuits shall be tested manually at least once per 123 days.

~~(2) On a STAGGERED TEST BASIS.~~

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area Monitor	SFCP	SFCP~~R~~	SFCP~~M~~	Note 1
b. Containment High Range	SFCP	SFCP~~R~~ Note 4	SFCP~~M~~	1, 2, 3, & 4
2. PROCESS MONITORS				
a. Containment Purge and Exhaust Isolation	Note 2	SFCP~~R~~	Note 3	5 & 6
b. Control Room Ventilation Intake Duct Monitors	SFCP	SFCP~~R~~	SFCP~~M~~ Note 6	Note 5
c. Main Steam Line Radiation Monitors	SFCP	SFCP~~R~~	SFCP~~M~~	1, 2, 3, & 4

Note 1 – With fuel in the spent fuel pool or building.

Note 2 – Within 8 hours prior to initiating containment purge operations and in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~ during containment purge operations.

Note 3 – Within 31 days prior to initiating containment purge operations and in accordance with the Surveillance Frequency Control Program~~at least once per 31 days~~ during containment purge operations.

Note 4 – Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG-0737.

Note 5 – MODES 1, 2, 3, 4, and during handling of irradiated fuel.

Note 6 - When the Control Room Ventilation Intake Duct Monitor is placed in an inoperable status solely for performance of this Surveillance, entry into associated ACTIONS may be delayed up to 3 hours.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Logarithmic Neutron Channel	SFCP~~M~~	N.A.
2.	Startup Channel	SFCP~~M~~	N.A.
3.	Reactor Trip Breaker Indication	SFCP~~M~~	N.A.
4.	Reactor Coolant Cold Leg Temperature	SFCP~~M~~	SFCP~~R~~
5.	Pressurizer Pressure	SFCP~~M~~	SFCP~~R~~
6.	Pressurizer Level	SFCP~~M~~	SFCP~~R~~
7.	Steam Generator Level	SFCP~~M~~	SFCP~~R~~
8.	Steam Generator Pressure	SFCP~~M~~	SFCP~~R~~
9.	Shutdown Cooling Flow Rate	SFCP~~M~~	SFCP~~R~~
10.	Condensate Storage Tank Level	SFCP~~M~~	SFCP~~R~~

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure (Normal Design Range)	SFCP~~M~~	SFCP~~R~~
2. Containment Pressure (High Range)	SFCP~~M~~	SFCP~~R~~
3. Pressurizer Pressure	SFCP~~M~~	SFCP~~R~~
4. Pressurizer Water Level	SFCP~~M~~	SFCP~~R~~
5. Steam Generator Pressure	SFCP~~M~~	SFCP~~R~~
6. Steam Generator Water Level	SFCP~~M~~	SFCP~~R~~
7. Refueling Water Tank Water Level	SFCP~~M~~	SFCP~~R~~
8. Containment Water Level – Wide Range	SFCP~~M~~	SFCP~~R~~
9. Emergency Feedwater Flow Rate	SFCP~~M~~	SFCP~~R~~
10. Reactor Coolant System Subcooling Margin Monitor	SFCP~~M~~	SFCP~~R~~
11. Pressurizer Safety Valve Acoustic Position Indication	SFCP~~M~~	SFCP~~R~~
12. Pressurizer Safety Valve Tail Pipe Temperature	SFCP~~M~~	SFCP~~R~~

TABLE 4.3-10 (con't)

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
13. In Core Thermocouples (Core-Exit Thermocouples)	SFCP~~M~~	SFCP~~R~~
14. Reactor Vessel Level Monitoring (RVLMS)	SFCP~~M~~	SFCP~~R~~

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2. *

ACTION:

With less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

* See Special Test Exception 3.10.3.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The reactor coolant loops listed below shall be in operable:

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

b. At least one of the above Reactor Coolant Loops shall be in operation.*

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops operable, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

SURVEILLANCE REQUIREMENTS

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

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

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

REACTOR COOLANT SYSTEM

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump.

3. Shutdown Cooling Loop (A) #.

4. Shutdown Cooling Loop (B) #.

b. At least one of the above coolant loops shall be in operation.*

APPLICABILITY: Modes 4 and 5.

ACTION:

a. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible and initiate action to make at least one steam generator available for decay heat removal via natural circulation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required shutdown cooling loop(s) shall be determined OPERABLE per the INSERVICE TESTING PROGRAM.

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

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be ≥ 23% indicated level in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

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

The normal or emergency power source may be inoperable in Mode 5.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume of ≤ 910 cubic feet (equivalent to ≤ 82% of wide range indicated level) and both pressurizer proportional heater groups shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

(a) With the pressurizer inoperable due to water volume ≥ 910 cubic feet, be in at least HOT SHUTDOWN with the reactor trip breakers open within 12 hours.

(b) With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters, either restore the inoperable emergency power supply in accordance with TS 3.8.1.1, Action b.3, for an inoperable Emergency Diesel Generator, or be in at least HOT SHUTDOWN within 12 hours.

(c) With the pressurizer inoperable due to a single proportional heater group having less than a 150 KW capacity, restore the inoperable proportional heater group to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within 12 hours.

(d) With the pressurizer inoperable due to both proportional heater groups being inoperable for any reason (Note 1), restore at least one proportional heater group to OPERABLE status within 24 hours, or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limits in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

4.4.4.2 The pressurizer proportional heater groups shall be determined to be OPERABLE.

(a) In accordance with the Surveillance Frequency Control Program~~At least once per 12 hours~~ by verifying emergency power is available to the heater groups, and

(b) In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by verifying that the summed power consumption of the two proportional heater groups is ≥ 150 KW.

Note 1: Action (d) is not applicable when the second group of required pressurizer heaters is intentionally made inoperable.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection instrumentation shall be demonstrated OPERABLE by:

a. Performing a CHANNEL CHECK of the required containment atmosphere radioactivity monitors in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

b. Performing a CHANNEL CHECK of the containment sump level monitor in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~.

c. Performing a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitors in accordance with the Surveillance Frequency Control Program~~at least once per 31 days~~.

d. Performing a CHANNEL CALIBRATION of the containment sump level monitor in accordance with the Surveillance Frequency Control Program~~at least once per 18 months~~.

e. Performing a CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitors in accordance with the Surveillance Frequency Control Program~~at least once per 18 months~~.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakage, except for primary to secondary leakage, shall be demonstrated to be within each of the above limits by:

a. Performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program~~at least once per 72 hours~~ during steady state operation except when operating in the shutdown cooling mode*.

b. Monitoring the reactor head flange leakoff temperature in accordance with the Surveillance Frequency Control Program~~at least once per 24 hours~~.

4.4.6.2.2 Primary to secondary leakage shall be verified to be ≤ 150 gallons per day through any one SG in accordance with the Surveillance Frequency Control Program~~at least once per 72 hours~~*.

4.4.6.2.3 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4.6-1 shall be demonstrated OPERABLE by individually verifying leakage to be within its limit:

a. Prior to entering MODE 2 after each refueling outage,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months, and

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

---

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

LIMITING CONDITION FOR OPERATION

3.4.8 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

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

ACTION

Note: The provisions of Specification 3.0.4.c are applicable to ACTION a and b.

a. With the DOSE EQUIVALENT I-131 not within limit:

1. Verify DOSE EQUIVALENT I-131 ≤ 60 μCi/gm once every 4 hours, and

2. Restore DOSE EQUIVALENT I-131 within limit within 48 hours.

b. With the DOSE EQUIVALENT XE-133 not within limit, restore DOSE EQUIVALENT XE-133 within limit within 48 hours.

c. With the requirements of ACTION a and/or b not met, or with DOSE EQUIVALENT I-131 > 60 μCi/gm, be in at least HOT STANDBY in 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1 Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 3100 μCi/gm in accordance with the Surveillance Frequency Control Program~~once every 7 days~~.*

4.4.8.2 Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm:*

a. in accordance with the Surveillance Frequency Control Program~~once every 14 days~~, and

b. between 2 and 6 hours after THERMAL POWER change of ≥ 15% RATED THERMAL POWER within a 1 hour period.

* Only required to be performed in MODE 1.

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program~~at least once per 30 minutes~~ during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in SAR Table 5.2-12. The results of these examinations shall be used to update Figures 3.4-2A, 3.4-2B and 3.4-2C.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one reactor coolant system vent path consisting of at least two valves in series shall be OPERABLE at each of the following locations:

1. Reactor Vessel Head

2. Pressurizer Steam Space (RCS High Point Vents)

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

ACTION:

a. With less than one vent path from each of the locations OPERABLE, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path(s) is maintained closed; restore the inoperable vent path to OPERABLE status within 30 days or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both vent paths 1 and 2 above inoperable, restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor coolant system vent path shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program~~at least once per 18 months~~ by verifying flow through the reactor coolant vent system vent paths.

4.4.12.1 Verify both sets of LTOP relief valve isolation valves are open in accordance with the Surveillance Frequency Control Program~~at least once per 72 hours~~ when the LTOP relief valves are being used for overpressure protection.

4.4.12.2 The RCS vent path shall be verified to be open in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~** when the vent path is being used for overpressure protection.

4.4.12.3 Verify that each SIT is isolated, when required, in accordance with the Surveillance Frequency Control Program~~once every 12 hours~~.

4.4.12.4 No additional LTOP relief valve Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

** Except when the vent path is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify this valve is open in accordance with the Surveillance Frequency Control Program~~at least once per 31 days~~.

SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 1413 and 1539 cubic feet (equivalent to an indicated level between 80.1% and 87.9%, respectively),

c. Between 2200 and 3000 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 624 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, due to boron concentration not within limits, restore the boron concentration to within limits within 72 hours, or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 700 psia within the next 12 hours.

b. With one safety injection tank inoperable due to inability to verify level or pressure, restore the SIT to OPERABLE status within 72 hours, or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 700 psia within the next 12 hours.

c. With one safety injection tank inoperable for reasons other than ACTION a or b, restore the SIT to OPERABLE status within 24 hours, or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 700 psia within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program~~At least once per 12 hours~~ by:

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

2. Verifying that each safety injection tank isolation valve (2CV-5003-1, 2CV-5023-1, 2CV-5043-2, and 2CV-5063-2) is open.

* With pressurizer pressure ≥ 700 psia.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ and within 6 hours after each solution volume increase of ≥ 5% of indicated tank level that is not the result of addition from the RWT, by verifying the boron concentration of the safety injection tank solution.

c. In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ when the RCS pressure is above 2000 psia, by verifying that power to the isolation valve operator is removed by maintaining the motor circuit breaker open under administrative control.

d. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

1. When the RCS pressure exceeds 700 psia, and

2. Upon receipt of a safety injection test signal.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program~~At least once per 12 hours~~ by verifying that the following valves are in the indicated positions with power to the 2CV-5101-1 and 2CV-5102-2 valve operators removed:

Valve Number	Valve Function	Valve Position
2CV-5101-1	HPSI Hot Leg Injection Isolation	Closed
2CV-5102-2	HPSI Hot Leg Injection Isolation	Closed
2BS-26	RWT Return Line	Open

b. In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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

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

2. At least once daily of the areas affected within containment if containment has been entered that day, and during the final entry when CONTAINMENT INTEGRITY is established.

d. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by a visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on SIAS and RAS test signals.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to the INSERVICE TESTING PROGRAM:

1. High-Pressure Safety Injection pump ≥ 1360.4 psid with 90 °F water.

2. Low-Pressure Safety Injection pump ≥ 156.25 psid with 90 °F water.

g. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

LPSI System
Valve Number

a. 2CV-5037-1

b. 2CV-5017-1

c. 2CV-5077-2

d. 2CV-5057-2

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

HPSI System – Single Pump

The sum of the injection line flow rates, excluding the highest flow rate is greater than or equal to 570 gpm.

LPSI System – Single Pump

a. Injection Leg 1, ≥ 1059 gpm

b. Injection Leg 2, ≥ 1059 gpm

c. Injection Leg 3, ≥ 1059 gpm

d. Injection Leg 4, ≥ 1059 gpm

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. An available borated water volume of between 384,000 and 503,300 gallons

b. Between 2500 and 3000 ppm of boron,

c. A minimum solution temperature of 40 °F, and

d. A maximum solution temperature of 110 °F

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program~~At least once per 7 days~~ by:

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

2. Verifying the boron concentration of the water.

b. In accordance with the Surveillance Frequency Control Program~~At least once per 24 hours~~ by verifying the RWT temperature.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ by verifying that each containment isolation manual valve and blind flange (Note 1) that is located outside containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative control as permitted by Specification 3.6.3.1.

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

c. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals in accordance with the Containment Leakage Rate Testing Program.

d. Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days by verifying each containment isolation manual valve and blind flange (Note 1) that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls as permitted by Specification 3.6.3.1.

Note 1: Valves and blind flanges in high radiation areas may be verified by use of administrative means.

SURVEILLANCE REQUIREMENTS

4.6.1.3.1 Each containment air lock shall be demonstrated OPERABLE as specified in the Containment Leakage Rate Testing Program[5,6].

4.6.1.3.2 Each containment air lock interlock shall be demonstrated OPERABLE by testing the air lock interlock mechanism in accordance with the Surveillance Frequency Control Program~~at least once per 184 days~~[7].

[5] Leakrate results shall also be evaluated against the acceptance criteria of specification 3.6.1.2.

[6] An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

[7] This surveillance requirement is only required to be performed upon entry or exit through the associated containment air lock.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE AND AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 The combination of containment internal pressure and average air temperature shall be maintained within the region of acceptable operation shown on Figure 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the point defined by the combination of containment internal pressure and average air temperature outside the region of acceptable operation shown on Figure 3.6-1, restore the combination of containment internal pressure and average air temperature to within the above limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure and average air temperature shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~. The containment average air temperature shall be the temperature of the air in the containment HVAC common return air duct upstream of the fan/cooler units.

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.6 The containment purge supply and exhaust isolation valves shall be closed and handswitch keys removed.

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

ACTION:

With one or more containment purge supply and/or exhaust isolation valves not closed with the handswitch keys removed, place the valve(s) in the closed position with handswitch keys(s) removed within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The containment purge supply and exhaust isolation valves shall be determined closed in accordance with the Surveillance Frequency Control Program~~at least once per 31 days~~.

3/4.6.2 DEPRESSURIZATION, COOLING, AND pH CONTROL SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWT on a Containment Spray Actuation Signal (CSAS) and automatically transferring suction to the containment sump on a Recirculation Actuation Signal (RAS). Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With both containment spray systems inoperable (Note 1):

1. Within 1 hour verify both CREVS trains are OPERABLE, and

2. Restore at least one containment spray system to OPERABLE status within 24 hours.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ by:

1. Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.

2. Verifying that the system piping is full of water from the RWT to at least elevation 505' (equivalent to > 12.5% indicated narrow range level) in the risers within the containment.

b. Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head when tested pursuant to the INSERVICE TESTING PROGRAM.

Note 1: ACTION b is not applicable when the second containment spray system is intentionally made inoperable.

SURVEILLANCE REQUIREMENTS (Continued)

c. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on CSAS and RAS test signals.

2. Verifying that upon a RAS test signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

3. Verifying that each spray pump starts automatically on a CSAS test signal.

d. Verify each spray nozzle is unobstructed following maintenance which could result in nozzle blockage.

CONTAINMENT SYSTEMS

CONTAINMENT SUMP BUFFERING AGENT

LIMITING CONDITION FOR OPERATION

3.6.2.2 The buffering agent baskets shall contain ≥ 308 ft^3 of sodium tetraborate (NaTB) decahydrate.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the buffering agent not within limits, restore the buffering agent to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and be in at least HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The buffering agent shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by verifying that the buffering agent baskets contain ≥ 308 ft^3 of NaTB decahydrate.

b. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by verifying that a sample from the buffering agent baskets provides adequate pH adjustment of borated water.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment cooling group shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program~~At least once per 14 days~~ by:

1. Verifying that service water flow rate to the group of cooling units is ≥ 1250 gpm and that each group has two operable fans.

2. Addition of a biocide to the service water during the surveillance in 4.6.2.3.a.1 above, whenever service water temperature is between 60 °F and 80 °F.

b. In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ by:

1. Starting (unless already operating) each operational cooling unit from the control room.

2. Verifying that each operational cooling unit operates for at least 15 minutes.

c. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by verifying that each cooling unit starts automatically on a CCAS test signal.

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program~~at least once per 18 months~~ by verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position.

4.6.3.1.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to the INSERVICE TESTING PROGRAM.

4.6.3.1.4 The containment purge supply and exhaust isolation valves shall be demonstrated OPERABLE as specified in the Containment Leakage Rate Testing Program.

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two emergency feedwater pumps and associated flow paths shall be OPERABLE with:

a. One motor driven pump capable of being powered from an OPERABLE emergency bus, and

b. One turbine driven pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

NOTE: Specification 3.0.4.b is not applicable.

With one emergency feedwater pump inoperable, restore the inoperable pump to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater pump shall be demonstrated OPERABLE:

a In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ by:

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

b. In accordance with the INSERVICE TESTING PROGRAM by:

1. Verifying the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head. This surveillance requirement is not required to be performed for the turbine driven EFW pump until 24 hours after exceeding 700 psia in the steam generators.

SURVEILLANCE REQUIREMENTS (Continued)

c In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on MSIS or EFAS test signals.

2. Verifying that the motor driven pump starts automatically upon receipt of an EFAS test signal.

3. Verifying that the turbine driven pump steam supply MOV opens automatically upon receipt of an EFAS test signal.

d. By verifying proper alignment of the required EFW flow paths by verifying flow from the condensate storage tank to each steam generator. This SR is required to be verified prior to entering MODE 2 whenever plant has been in MODES 4, 5, 6, or defueled for > 30 days.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 At least one condensate storage tank (CST) shall be OPERABLE with a minimum contained water volume of either:

a. 160,000 gallons in either 2T41A or 2T41B, or

b. A minimum of 267,000 gallons of water is available in condensate storage tank, T41B, when required for both units. A minimum of 160,000 gallons of water is available in T41B when only required for Unit 2.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the required condensate storage tank inoperable, within 4 hours either:

a. Restore at least one CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the service water system as a backup supply to the emergency feedwater pumps and restore at least one condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The above required condensate storage tank shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~ by verifying the contained water volume is within its limits when the tank is the supply source for the emergency feedwater pumps.

4.7.1.3.2 The service water system shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~ by verifying that at least one service water loop is operating and that the service water system - emergency feedwater system isolation valves are either open or OPERABLE whenever the service water system is the supply source for the emergency feedwater pumps.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

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

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 90 °F when the pressure of either coolant in the steam generator is > 275 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to ≤ 275 psig within 30 minutes, and

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

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generators shall be determined to be < 275 psig in accordance with the Surveillance Frequency Control Program~~at least once per hour~~ when the temperature of either the primary or secondary coolant is < 90 °F.

3/4.7.3 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

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

a. In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on CCAS, MSIS and RAS test signals.

PLANT SYSTEMS

3/4.7.4 EMERGENCY COOLING POND

LIMITING CONDITION FOR OPERATION

3.7.4.1 The emergency cooling pond (ECP) shall be OPERABLE with:

a. A minimum contained water volume of 70 acre-feet.

b. An average water temperature of ≤ 100 °F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the volume and/or temperature requirements of the above specification not satisfied or, with the requirements of Action b not met, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. If degradation is noted pursuant to 4.7.4.1.d below or by other inspection, perform an evaluation to determine that the ECP remains acceptable for continued operation within 7 days.

SURVEILLANCE REQUIREMENTS

4.7.4.1 The ECP shall be determined OPERABLE:

a. In accordance with the Surveillance Frequency Control Program~~At least once per 24 hours~~ by verifying that the indicated water level of the ECP is greater than or equal to that required for an ECP volume of 70 acre-feet.

b. In accordance with the Surveillance Frequency Control Program~~At least once per 24 hours~~ during the period of June 1 through September 30 by verifying that the pond's average water temperature at the point of discharge from the pond is within its limit.

c. In accordance with the Surveillance Frequency Control Program~~At least once per 12 months~~ by making soundings of the pond and verifying:

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.

d. In accordance with the Surveillance Frequency Control Program~~At least once per 12 months~~ by performance of a visual inspection of the ECP to verify conformance with design requirements.

3/4.7.5 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.5.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Dardanelle Reservoir exceeds 350 feet Mean Sea Level USGS datum, at the intake structure.

APPLICABILITY: When a flood warning exists at the facility site.

ACTION:

With the water level at the intake structure above elevation 350 feet Mean Sea Level USGS datum, initiate and complete within 4 hours, closure of the openings and penetrations listed in Table 3.7-6 using the equipment listed in Table 3.7-6.

SURVEILLANCE REQUIREMENTS

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

a. Measurement in accordance with the Surveillance Frequency Control Program~~at least once per 24 hours~~ when the water level is below elevation 350 feet Mean Sea Level USGS datum, and

b. Measurement in accordance with the Surveillance Frequency Control Program~~at least once per 2 hours~~ when the water level is equal to or above elevation 350 feet Mean Sea Level USGS datum.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 Each control room emergency air conditioning system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ by:

1. Starting each unit from the control room, and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤ 84 °F D.B.

b. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by verifying a system flow rate of 9900 cfm ± 10%.

4.7.6.1.2 Each control room emergency air filtration system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ by verifying that the system operates for at least 15 minutes.

b. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by verifying that on a control room high radiation signal, either actual or simulated, the system automatically isolates the control room and switches into a recirculation mode of operation.

c. By performing the required Control Room Emergency Ventilation filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

d. Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of ≥ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

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

1. Either decontaminated and repaired, or

2. Disposed of in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

4.7.9.1.2 Test Frequencies – Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequencies described below.

a. Sources in use – In accordance with the Surveillance Frequency Control Program~~At least once per six months~~ for all sealed sources containing radioactive material:

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program~~at least once per 7 days~~ by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program~~at least once per 18 months~~ during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE: (Note 1)

a. In accordance with the Surveillance Frequency Control Program~~At least once per 31 days on a STAGGERED TEST BASIS~~ by:

1. Verifying the fuel level in the day fuel tank.

2. deleted

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts from a standby condition and accelerates to at least 900 rpm in $\leq$ 15 seconds. (Note 2)

5. Verifying the generator is synchronized, loaded to an indicated 2600 to 2850 Kw and operates for $\geq$ 60 minutes. (Notes 3 & 4)

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

b. deleted

Note 1
All planned diesel generator starts for the purposes of these surveillances may be preceded by prelube procedures.

Note 2
This diesel generator start from a standby condition in $\leq$ 15 sec. shall be accomplished at least once every 184 days. All other diesel generator starts for this surveillance may be in accordance with vendor recommendations.

Note 3
Diesel generator loading may be accomplished in accordance with vendor recommendations such as gradual loading.

Note 4
Momentary transients outside this load band due to changing loads will not invalidate the test. Load ranges are allowed to preclude over- loading the diesel generators.

SURVEILLANCE REQUIREMENTS (Continued)

c. In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by:

1. Deleted

2. Verifying during shutdown that the automatic sequence time delay relays are OPERABLE at their setpoint ± 10% of the elapsed time for each load block.

3. Verifying during shutdown the generator capability to reject a load of greater than or equal to its associated single largest post-accident load, and maintain voltage at 4160 ± 500 volts and frequency at 60 ± 3 Hz.

4. Verifying during shutdown the generator capability to reject a load of 2850 Kw without exceeding 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal, whichever is lower.

5. Simulating during shutdown a loss of offsite power by itself, and:

   a. Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b. Verifying the diesel starts from a standby condition on the undervoltage auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected shutdown loads through the time delay relays and operates for ≥ 5 minutes while its generator is loaded with the shutdown loads.

6. Verifying during shutdown that on a Safety Injection Actuation Signal (SIAS) actuation test signal (without loss of offsite power) the diesel generator starts on the auto-start signal and operates on standby for ≥ 5 minutes.

11. Verifying during shutdown the diesel generator's capability to:

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

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

    c) Proceed through its shutdown sequence.

12. Verifying during shutdown that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the auto-connected emergency (accident) loads with offsite power.

13. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

d. In accordance with the Surveillance Frequency Control Program~~At least once per 10 years~~ or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 900 rpm in ≤ 15 seconds.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.8.1.3 The stored diesel fuel oil shall be within limits for each required diesel generator.

APPLICABILITY: When associated diesel generator is required to be OPERABLE.

ACTION:

With the volume of the stored diesel fuel oil less than 22,500 gallons for either fuel oil storage tank or the new or stored fuel oil properties outside the limits of the Diesel Fuel Oil Testing Program, perform the following as appropriate: (Note – Separate ACTION entry is allowed for each diesel generator.)

1. If one or more fuel storage tanks contain less than 22,500 gallons and greater than 17,446 gallons, restore the fuel oil volume to within limits within 48 hours.

2. If the stored fuel oil total particulates are not within limits for one or more diesel generators, restore fuel oil total particulates to within limits within 7 days.

3. If new fuel oil properties are not within limits for the one or more diesel generators, restore stored fuel oil properties to within limits within 30 days.

4. If ACTION 1 is not met within the allowable outage time or is outside the allowable limits, or if ACTION 2 or 3 is not met within the allowable outage time, then immediately declare the associated diesel generator inoperable.

SURVEILLANCE REQUIREMENTS

4.8.1.3.1 In accordance with the Surveillance Frequency Control Program~~At least once per 31 days on a STAGGERED TEST BASIS~~ verify the fuel oil storage tank contains ≥ 22,500 gallons of fuel.

4.8.1.3.2 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.

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION – OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized with tie breakers open between redundant busses:

4160 volt Emergency Bus # 2A3

4160 volt Emergency Bus # 2A4

480 volt Emergency Bus # 2B5

480 volt Emergency Bus # 2B6

120 volt A.C. Vital Bus # 2RS1

120 volt A.C. Vital Bus # 2RS2

120 volt A.C. Vital Bus # 2RS3

120 volt A.C. Vital Bus # 2RS4

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE with tie breakers open between redundant busses in accordance with the Surveillance Frequency Control Program~~at least once per 7 days~~ by verifying correct breaker alignment and indicated power availability.

ELECTRICAL POWER SYSTEMS

A.C. DISTRIBUTION – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses shall be OPERABLE:

1 - 4160 volt Emergency Bus

1 - 480 volt Emergency Load Center Bus

4 - 480 volt Motor Control Center Busses

2 - 120 volt A.C. Vital Busses

APPLICABILITY: MODES 5 and 6

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, immediately suspend core alterations, the movement of irradiated fuel assemblies, and any operations involving positive reactivity additions.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program~~at least once per 7 days~~ by verifying correct breaker alignment and indicated power availability.

DC SOURCES – OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required full capacity chargers inoperable:

   i. Restore the battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours, and

   ii. Verify battery float current ≤ 2 amps once per 12 hours.

b. With one DC electrical power subsystem inoperable for reasons other than ACTION ‘a’ above, restore the inoperable DC electrical power subsystem to OPERABLE status within 2 hours.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 In accordance with the Surveillance Frequency Control Program~~At least once per 7 days~~ by verifying that the battery terminal voltage is greater than or equal to the minimum established float voltage.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.2.3.2 In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by verifying that each battery charger supplies ≥ 300 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours or, by verifying that 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.

4.8.2.3.3 In accordance with the Surveillance Frequency Control Program~~At least once per 18 months~~ by verifying that the battery capacity is adequate to supply, and maintain in OPERABLE status, required emergency loads for the design duty cycle when subjected to a battery service test. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this Surveillance. The battery performance discharge test required by Surveillance Requirement 4.8.3.6 may be performed in lieu of the battery service test once per 60 months.

DC SOURCES – SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

1 - 125-volt DC bus, and

1 - 125-volt battery bank and charger supplying the above DC bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the required battery charger inoperable:

i. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours, and

ii. Verify battery float current ≤ 2 amps once per 12 hours.

b. With the requirements of ACTION ‘a’ not met or with the above complement of DC equipment and bus otherwise inoperable, immediately suspend core alterations, the movement of irradiated fuel assemblies, and any operations involving positive reactivity additions.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program~~at least once per 7 days~~ by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirements 4.8.2.3.1, 4.8.2.3.2, and 4.8.2.3.3; however, while each of these Surveillance Requirements must be met, Surveillance Requirements 4.8.2.3.2 and 4.8.2.3.3 are not required to be performed.

SURVEILLANCE REQUIREMENTS

4.8.3.1 In accordance with the Surveillance Frequency Control Program~~At least once per 7 days~~ by verifying that each battery float current is ≤ 2 amps. This Surveillance is not required when battery terminal voltage is less than the minimum established float voltage of Surveillance Requirement 4.8.2.3.1.

4.8.3.2 In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ by verifying that each battery pilot cell float voltage is ≥ 2.07 V.

4.8.3.3 In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ by verifying that each battery connected cell electrolyte level is greater than or equal to minimum established design limits.

4.8.3.4 In accordance with the Surveillance Frequency Control Program~~At least once per 31 days~~ by verifying that each battery pilot cell temperature is greater than or equal to minimum established design limits.

4.8.3.5 In accordance with the Surveillance Frequency Control Program~~At least once per 92 days~~ by verifying that each battery connected cell float voltage is ≥ 2.07 V.

4.8.3.6 In accordance with the Surveillance Frequency Control Program~~At least once per 60 months~~ by verifying the battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this Surveillance. In addition~~ to the 60-month test interval~~, the performance discharge test shall be performed:

a. At least once per 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating, and

b. At least once per 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating.

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of the reactor coolant and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of $\geq$ 2500 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at $\geq$ 40 gpm of $\geq$ 2500 ppm boric acid solution until K_{eff} is reduced to $\leq$ 0.95 or the boron concentration is restored to $\geq$ 2500 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant and the refueling canal shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program~~at least once per 72 hours~~.

---

* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable, determine the boron concentration of the reactor coolant system at least once per 12 hours.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. A CHANNEL CHECK in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~,

b. A CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program~~at least once per 7 days~~, and

c. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS.

CONTAINMENT BUILDING PENETRATION

LIMITING CONDITION FOR OPERATION

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

a. The equipment door is capable* of being closed,

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

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

1. Closed* by a manual or automatic isolation valve, blind flange, or equivalent, or

2. Capable* of being closed by an OPERABLE containment purge and exhaust isolation system.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment penetrations shall be determined to be in its above required conditions within 72 hours prior to the start of and in accordance with the Surveillance Frequency Control Program~~at least once per 7 days~~ during CORE ALTERATIONS or movement of irradiated fuel in the containment.

* Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. Administrative controls shall ensure that appropriate personnel are aware that when containment penetrations, including both personnel airlock doors and/or the equipment door are open, a specific individual(s) is designated and available to close the penetration following a required evacuation of containment, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door and/or the equipment door be capable of being quickly removed.

Next page is 3/4 9-6

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and in accordance with the Surveillance Frequency Control Program~~at least once per 12 hours~~ during CORE ALTERATIONS.

CRANE TRAVEL – SPENT FUEL POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pool.

APPLICABILITY: With fuel assemblies in the spent fuel pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 The crane electrical power disconnect which prevents crane travel over the spent fuel pool shall be verified open under administrative control in accordance with the Surveillance Frequency Control Program~~at least once per 7 days~~, or the crane travel interlock which prevents crane travel over the spent fuel pool shall be demonstrated OPERABLE within 4 hours prior to each use of the crane for lifting loads in excess of 2000 pounds.

SHUTDOWN COOLING AND COOLANT CIRCULATION

SHUTDOWN COOLING – ONE LOOP

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

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

b. The shutdown cooling loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 A shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of ≥ 2000 gpm in accordance with the Surveillance Frequency Control Program~~at least once per 24 hours~~.

REFUELING OPERATIONS

WATER LEVEL – REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel while in MODE 6, except during latching and unlatching of CEAs.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.9 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and in accordance with the Surveillance Frequency Control Program~~at least once per 24 hours~~ thereafter during movement of fuel assemblies or CEAs.

SPENT FUEL POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.10 The water level in the spent fuel pool shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program~~at least once per 7 days~~ when irradiated fuel assemblies are in the spent fuel pool.

Next Page is 3/4 9-14

REFUELING OPERATIONS

FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12.a Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.95 w/o U-235. The provisions of Specification 3.0.3 are not applicable.

3.9.12.b Storage in the spent fuel pool shall be further restricted by the limits specified in Table 3.9-1. The provisions of Specification 3.0.3 are not applicable.

3.9.12.c The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 2000 parts per million.

APPLICABILITY: During storage of fuel in the spent fuel pool

ACTION:

Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined a fuel assembly has been placed in an incorrect location until such time as the correct storage location is determined. Move the assembly to its correct location before resumption of any other fuel movement.

Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined the pool boron concentration is less than 2001 ppm, until such time as the boron concentration is increased to 2001 ppm or greater.

SURVEILLANCE REQUIREMENTS

4.9.12.a Verify all fuel assemblies to be placed in the spent fuel pool have an initial enrichment of less than or equal to 4.95 w/o U-235 by checking the assemblies' design documentation.

4.9.12.b Verify all fuel assemblies to be placed in the spent fuel pool are within the limits of Table 3.9-1 by checking the assemblies' design and burnup documentation.

4.9.12.c Verify in accordance with the Surveillance Frequency Control Program~~at least once per 31 days~~ the spent fuel pool boron concentration is greater than 2000 ppm.

4.9.12.d Verify Metamic properties are in accordance with, and are maintained within the limits of, the Metamic Coupon Sampling Program.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

a. With any CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each CEA required either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program~~at least once per 2 hours~~.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of Functional Unit 14 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The linear heat rate limit shall be maintained by either:

1. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

2. Operating within the region of acceptable operation as specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service.)

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With any of the above limits being exceeded while any of the above requirements are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of the above Specification, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program~~at least once per hour~~ during PHYSICS TESTS in which any of the above requirements are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within its limits during PHYSICS TESTS above 5% of RATED THERMAL POWER in which any of the above requirements are suspended.

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

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

b. The reactor trip setpoints of the OPERABLE power level channels are set at ≤ 20% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program~~at least once per hour~~ during startup and PHYSICS TESTS.

4.10.3.2 Each wide range logarithmic and power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program~~at least once per hour~~ during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the incore detection system during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

SPECIAL TEST EXCEPTIONS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.5 The minimum temperature for criticality limits of Specification 3.1.1.5 may be suspended during low temperature PHYSICS TESTS provided:

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

b. The reactor trip setpoints on the OPERABLE Linear Power Level - High neutron flux monitoring channels are set at ≤ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation shown on Figure 3.4-2.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

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

b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figure 3.4-2, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the engineering evaluation required by Specification 3.4.9.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The Reactor Coolant System temperature and pressure relationship shall be verified to be within the acceptable region for operation of Figure 3.4-2 in accordance with the Surveillance Frequency Control Program~~at least once per hour~~.

4.10.5.2 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program~~at least once per hour~~.

4.10.5.3 Each Logarithmic Power Level and Linear Power Level channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1 The quantity of radioactive material contained in each unprotected outside temporary radioactive liquid storage tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material exceeding the above limit, immediately suspend all additions of radioactive material to the affected tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluents Release Report pursuant to Specification 6.9.3.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1 The quantity of radioactive material contained in each unprotected outside temporary radioactive liquid storage tank shall be determined to be within the above limit by analyzing a representative sample of the contents of the tank in accordance with the Surveillance Frequency Control Program~~at least once per 7 days~~ when radioactive materials are being added to the tank.

---

* Tanks included in this specification are those outdoor temporary tanks that 1) are not surrounded by liners, dikes, or walls capable of holding the tank contents, and 2) do not have overflows and surrounding area drains connected to the liquid radwaste treatment system.

3/4.11.2 GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 82,400 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluent Release Report pursuant to Specification 6.9.3.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit in accordance with the Surveillance Frequency Control Program~~at least once per 24 hours~~ when radioactive materials are being added to the tank and the reactor coolant activity exceeds the limits of Specification 3.4.8.

6.5 PROGRAMS AND MANUALS

6.5.2 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~~least once per 18 months~~. The provisions of Surveillance Requirements 4.0.2 are applicable.

6.5.3 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.

6.5.4 Radioactive Effluent Controls Program

This program conforms with 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 to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS, conforming to 10 CFR 50, Appendix I;

6.5.12 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 inleakage 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~~of one train every 18 months~~. The results shall be trended and used as part of the ~~18 month~~ assessment of the CRE boundary.

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

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

6.5.13 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 4.0.2 and SR 4.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance frequencies.

6.5.14 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 require 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.

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

d. Proposed changes that do not meet the criteria of 6.5.14b above 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).

6.5.17 Metamic Coupon Sampling Program

A coupon surveillance program will be implemented to maintain surveillance of the Metamic absorber material under the radiation, chemical, and thermal environment of the SFP. The purpose of the program is to establish the following:

- Coupons will be examined on a two year basis for the first three intervals with the first coupon retrieved for inspection being on or before October 31, 2009 and thereafter at increasing intervals over the service life of the inserts.

- Measurements to be performed at each inspection will be as follows:

  a. Physical observations of the surface appearance to detect pitting, swelling or other degradation,

  b. Length, width, and thickness measurements to monitor for bulging and swelling

  c. Weight and density to monitor for material loss, and

  d. Neutron attenuation to confirm the B-10 concentration or destructive chemical testing to determine the boron content.

- The provisions of SR 4~~3~~.0.2 are applicable to the Metamic Coupon Sampling Program.

- The provisions of SR 4~~3~~.0.3 are not applicable to the Metamic Coupon Sampling Program.

6.5.18 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.

ATTACHMENT 4 to

2CAN021802

REVISED TECHNICAL SPECIFICATION PAGES

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or controlled leakage.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary to secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

AZIMUTHAL POWER TILT – T_q

1.17 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

DOSE EQUIVALENT I-131

1.18 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same committed effective dose equivalent (CEDE) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The CEDE dose conversion factors used to determine the DOSE EQUIVALENT I-131 shall be performed using Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

DOSE EQUIVALENT XE-133

1.19 DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

1.20 Deleted

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	> 5%	≥ 300 °F
2. STARTUP	≥ 0.99	$\leq 5\%$	≥ 300 °F
3. HOT STANDBY	< 0.99	0	≥ 300 °F
4. HOT SHUTDOWN	< 0.99	0	300 °F > T_{avg} > 200 °F
5. COLD SHUTDOWN	< 0.99	0	≤ 200 °F
6. REFUELING**	≤ 0.95	0	≤ 140 °F

* Excluding decay heat.

** Reactor vessel head unbolted or removed and fuel in the vessel.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S/U	Prior to each reactor startup
N.A.	Not applicable
SFCP	In accordance with the Surveillance Frequency Control Program

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN – T_{avg} > 200 °F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With the SHUTDOWN MARGIN less than that required above, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

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

b. When in MODES 1 or 2#, in accordance with the Surveillance Frequency Control Program by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of (e) below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, in accordance with the Surveillance Frequency Control Program by consideration of at least the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% $\Delta k/k$ in accordance with the Surveillance Frequency Control Program. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

REACTIVITY CONTROL SYSTEMS

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than that required above, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

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

b. In accordance with the Surveillance Frequency Control Program by consideration of at least the following factors:

   1. Reactor coolant system boron concentration,
   2. CEA position,
   3. Reactor coolant system average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration, and
   6. Samarium concentration.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be $\geq$ 2000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the reactor coolant system < 2000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be $\geq$ 2000 gpm within one hour prior to the start of and in accordance with the Surveillance Frequency Control Program during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump or containment spray pump is in operation as a shutdown cooling pump and supplying $\geq$ 2000 gpm through the reactor coolant system.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 540 °F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 540 °F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be: $\geq$ 540 °F in accordance with the Surveillance Frequency Control Program.

With $K_{eff} \geq 1.0$.

* See Special Test Exception 3.10.5.

Next page is 3/4 1-17

ACTION: (Continued)

e. With more than one CEA misaligned from any other CEA in its group by more than 7 inches (indicated position), be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group in accordance with the Surveillance Frequency Control Program.

4.1.3.1.2 Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction in accordance with the Surveillance Frequency Control Program.

Note 1 - Movement of CEA 4 is not required for the remainder of Cycle 26. If an outage of sufficient duration occurs prior to the end of Cycle 26, maintenance activities will be performed to restore the CEA.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5 inches of each other in accordance with the Surveillance Frequency Control Program.

POSITION INDICATOR CHANNELS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each CEA not fully inserted.

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

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.

* With the reactor trip breakers in the closed position.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual CEA drop time, from a fully withdrawn position, shall be ≤ 3.7 seconds and the arithmetic average of the CEA drop times of all CEAs, from a fully withdrawn position, shall be ≤ 3.2 seconds from when the electrical power is interrupted to the CEA drive mechanisms until the CEAs reach their 90 percent insertion positions with:

a. T_{avg} ≥ 525 °F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the CEA drop times determined to exceed either of the above limits, restore the CEA drop times to within the above limits prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of all CEAs shall be demonstrated through measurement prior to reactor criticality:

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

b. For specifically affected individuals CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. In accordance with the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to the Full Out position.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than the Full Out position, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Withdraw the CEA to the Full Out position, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to the Full Out position:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. In accordance with the Surveillance Frequency Control Program.

---

* See Special Test Exception 3.10.2.

With $K_{eff} \geq 1.0$.

ACTION: (Continued)

b) Both CEACs inoperable:

Be in at least HOT STANDBY within 8 hours of exceeding the Transient Insertion Limit.

b. With the regulating CEA groups or Group P CEAs inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, either:

1. Restore the regulating groups or Group P CEAs to within the Long Term Steady State Insertion Limit within two hours, or

2. Be in at least HOT STANDBY within the next 6 hours.

c. With the regulating CEA groups or Group P CEAs inserted between the Short Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 4 hours per 24 hour interval, operation may proceed provided any subsequent increase in thermal power is restricted to ≤ 5% of rated thermal power per hour.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group and Group P CEAs shall be determined to be within the Transient Insertion Limits in accordance with the Surveillance Frequency Control Program except during time intervals when the PDIL Alarm is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups or Group P CEAs are inserted beyond the Long Term Steady State Insertion Limit or the Short Term Steady State Insertion Limit but within the Transient Insertion Limit shall be determined in accordance with the Surveillance Frequency Control Program.

3/4. 2 POWER DISTRIBUTION LIMITS

3/4.2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate limit shall be maintained by either:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

b. Operating within the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT using any operable CPC Channel (when COLSS is out of service).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

a. With COLSS in service and the linear heat rate limit not being maintained as indicated by COLSS calculated core power exceeding the COLSS calculated core power operating limit based on linear heat rate, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limit and either:

   1. Restore the linear heat rate to within its limits within 1 hour of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

b. With COLSS out of service and the linear heat rate limit not being maintained as indicated by operation outside the region of acceptable operation specified in the CORE OPERATING LIMITS REPORT, either:

   1. Restore the linear heat rate to within its limits within 2 hours of the initiating event, or

   2. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying in accordance with the Surveillance Frequency Control Program that the linear heat rate, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.

4.2.1.3 In accordance with the Surveillance Frequency Control Program, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on linear heat rate.

POWER DISTRIBUTION LIMITS

RADIAL PEAKING FACTORS

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*

ACTION:

With a F^m_{xy} exceeding a corresponding F^c_{xy}, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to PLANAR RADIAL PEAKING FACTOR by a factor equivalent to $\geq F^m_{xy} / F^c_{xy}$ and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0] \times 100\%$ is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}); or

c. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}), obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. In accordance with the Surveillance Frequency Control Program in MODE 1.

* See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS

4.2.3 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt in accordance with the Surveillance Frequency Control Program when the COLSS is inoperable.

c. Verifying in accordance with the Surveillance Frequency Control Program, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors in accordance with the Surveillance Frequency Control Program to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying in accordance with the Surveillance Frequency Control Program that the DNBR, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.

4.2.4.3 In accordance with the Surveillance Frequency Control Program, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 120.4 x 10^6 lbm/hr.

APPLICABILITY: MODE 1

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program.

POWER DISTRIBUTION LIMITS

REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The Reactor Coolant Cold Leg Temperature (T_c) shall be maintained between 542 °F and 554.7 °F.

APPLICABILITY: MODE 1 above 30% of RATED THERMAL POWER.

ACTION:

With the Reactor Coolant Cold Leg Temperature exceeding its limit, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The Reactor Coolant Cold Leg Temperature shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program.

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limits in accordance with the Surveillance Frequency Control Program using the COLSS or any OPERABLE Core Protection Calculator channel.

* See Special Test Exception 3.10.2.

PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The average pressurizer pressure shall be maintained between 2025 psia and 2275 psia.

APPLICABILITY: MODE 1.

ACTION:

With the average pressurizer pressure exceeding its limits, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The average pressurizer pressure shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Neutron detectors are exempt from response time testing.

4.3.1.1.4 The Core Protection Calculator System shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

4.3.1.1.5 The affected Core Protection Calculator Channel shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a valid CPC Cabinet High Temperature alarm.

TABLE 4.3-1

REACTOR PROTECTION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U (1)	N.A.
2. Linear Power Level – High	SFCP	SFCP (2,3,4)	SFCP	1,2
3. Logarithmic Power Level – High	SFCP	SFCP (4)	SFCP S/U (1)	1,2,3*,4*,5*
4. Pressurizer Pressure – High	SFCP	SFCP	SFCP	1,2
5. Pressurizer Pressure – Low	SFCP	SFCP	SFCP	1,2
6. Containment Pressure – High	SFCP	SFCP	SFCP	1,2
7. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1,2
8. Steam Generator Level – Low	SFCP	SFCP	SFCP	1,2
9. Local Power Density – High	SFCP	SFCP (2,4,5)	SFCP (6)	1,2
10. DNBR – Low	SFCP	SFCP (2,4,5,7)	SFCP (6)	1,2
11. Reactor Protection System Logic	N.A.	N.A.	SFCP	1,2,3*,4*,5*
12. Reactor Trip Breakers	N.A.	N.A.	SFCP	1,2,3*,4*,5*
13. Core Protection Calculators	SFCP	SFCP (2,4,5)	SFCP (6,9)	1,2
14. CEA Calculators	SFCP	SFCP	SFCP (6)	1,2

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and, if necessary, adjust the CPC flow calibration addressable constant FC1 such that each CPC indicated flow is less than or equal to the measured flow rate.

(8) - Deleted

(9) - The CPC CHANNEL FUNCTIONAL TEST shall include the verification that the correct values of addressable constants are installed in each OPERABLE CPC.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNELS CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Containment Pressure – High	SFCP	SFCP	SFCP	1,2,3
c. Pressurizer Pressure – Low	SFCP	SFCP	SFCP	1,2,3
d. Automatic Actuation Logic	N.A.	N.A.	SFCP (1)	1,2,3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Containment Pressure -- High - High	SFCP	SFCP	SFCP	1,2,3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP (1)	1,2,3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Containment Pressure -- High	SFCP	SFCP	SFCP	1,2,3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP (1)	1,2,3
4. MAIN STEAM AND FEEDWATER ISOLATION (MSIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Steam Generator Pressure – Low	SFCP	SFCP	SFCP	1,2,3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP (1)	1,2,3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNELS CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. CONTAINMENT COOLING (CCAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	N.A.
b. Containment Pressure – High	SFCP	SFCP	SFCP	1,2,3
c. Pressurizer Pressure – Low	SFCP	SFCP	SFCP	1,2,3
d. Automatic Actuation Logic	N.A.	N.A.	SFCP (1)	1,2,3
6. RECIRCULATION (RAS)				
a. Manual (Trip Buttons) (a)	N.A.	N.A.	SFCP	N.A.
b. Refueling Water Tank – Low	SFCP	SFCP	SFCP	1,2,3
c. Automatic Actuation Logic	N.A.	N.A.	SFCP (1)	1,2,3
7. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	SFCP	SFCP	SFCP	1,2,3
b. 460 volt Emergency Bus Undervoltage (Degraded Voltage)	SFCP	SFCP	SFCP	1,2,3
8. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Switches)	N.A.	N.A.	SFCP	N.A.
b. SG Level and Pressure (A/B) – low and ΔP (A/B) – High	SFCP	SFCP	SFCP	1,2,3
c. SG Level (A/B) – Low and No Pressure – Low Trip (A/B)	SFCP	SFCP	SFCP	1,2,3
d. Automatic Actuation Logic	N.A.	N.A.	SFCP (1)	1,2,3

Table 4.3-2 (Continued)

TABLE NOTATION

(a) Remote manual not provided for RAS. These are local manuals at each ESF auxiliary relay cabinet.

(1) The logic circuits shall be tested manually at least once per 123 days.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area Monitor	SFCP	SFCP	SFCP	Note 1
b. Containment High Range	SFCP	SFCP Note 4	SFCP	1, 2, 3, & 4
2. PROCESS MONITORS				
a. Containment Purge and Exhaust Isolation	Note 2	SFCP	Note 3	5 & 6
b. Control Room Ventilation Intake Duct Monitors	SFCP	SFCP	SFCP Note 6	Note 5
c. Main Steam Line Radiation Monitors	SFCP	SFCP	SFCP	1, 2, 3, & 4

Note 1 – With fuel in the spent fuel pool or building.

Note 2 – Within 8 hours prior to initiating containment purge operations and in accordance with the Surveillance Frequency Control Program during containment purge operations.

Note 3 – Within 31 days prior to initiating containment purge operations and in accordance with the Surveillance Frequency Control Program during containment purge operations.

Note 4 – Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG-0737.

Note 5 – MODES 1, 2, 3, 4, and during handling of irradiated fuel.

Note 6 - When the Control Room Ventilation Intake Duct Monitor is placed in an inoperable status solely for performance of this Surveillance, entry into associated ACTIONS may be delayed up to 3 hours.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Logarithmic Neutron Channel	SFCP	N.A.
2.	Startup Channel	SFCP	N.A.
3.	Reactor Trip Breaker Indication	SFCP	N.A.
4.	Reactor Coolant Cold Leg Temperature	SFCP	SFCP
5.	Pressurizer Pressure	SFCP	SFCP
6.	Pressurizer Level	SFCP	SFCP
7.	Steam Generator Level	SFCP	SFCP
8.	Steam Generator Pressure	SFCP	SFCP
9.	Shutdown Cooling Flow Rate	SFCP	SFCP
10.	Condensate Storage Tank Level	SFCP	SFCP

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure (Normal Design Range)	SFCP	SFCP
2. Containment Pressure (High Range)	SFCP	SFCP
3. Pressurizer Pressure	SFCP	SFCP
4. Pressurizer Water Level	SFCP	SFCP
5. Steam Generator Pressure	SFCP	SFCP
6. Steam Generator Water Level	SFCP	SFCP
7. Refueling Water Tank Water Level	SFCP	SFCP
8. Containment Water Level – Wide Range	SFCP	SFCP
9. Emergency Feedwater Flow Rate	SFCP	SFCP
10. Reactor Coolant System Subcooling Margin Monitor	SFCP	SFCP
11. Pressurizer Safety Valve Acoustic Position Indication	SFCP	SFCP
12. Pressurizer Safety Valve Tail Pipe Temperature	SFCP	SFCP

TABLE 4.3-10 (con't)

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
13. Containment Pressure (Normal Design Range)	SFCP	SFCP
14. Reactor Vessel Level Monitoring (RVLMS)	SFCP	SFCP

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2. *

ACTION:

With less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

---

* See Special Test Exception 3.10.3.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The reactor coolant loops listed below shall be in operable:

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

b. At least one of the above Reactor Coolant Loops shall be in operation.*

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops operable, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

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

REACTOR COOLANT SYSTEM

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump.
2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump.
3. Shutdown Cooling Loop (A) #.
4. Shutdown Cooling Loop (B) #.

b. At least one of the above coolant loops shall be in operation.*

APPLICABILITY: Modes 4 and 5.

ACTION:

a. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible and initiate action to make at least one steam generator available for decay heat removal via natural circulation. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required shutdown cooling loop(s) shall be determined OPERABLE per the INSERVICE TESTING PROGRAM.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be ≥ 23% indicated level in accordance with the Surveillance Frequency Control Program.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

---

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

\# The normal or emergency power source may be inoperable in Mode 5.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume of ≤ 910 cubic feet (equivalent to ≤ 82% of wide range indicated level) and both pressurizer proportional heater groups shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

(a) With the pressurizer inoperable due to water volume ≥ 910 cubic feet, be in at least HOT SHUTDOWN with the reactor trip breakers open within 12 hours.

(b) With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters, either restore the inoperable emergency power supply in accordance with TS 3.8.1.1, Action b.3, for an inoperable Emergency Diesel Generator, or be in at least HOT SHUTDOWN within 12 hours.

(c) With the pressurizer inoperable due to a single proportional heater group having less than a 150 KW capacity, restore the inoperable proportional heater group to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within 12 hours.

(d) With the pressurizer inoperable due to both proportional heater groups being inoperable for any reason (Note 1), restore at least one proportional heater group to OPERABLE status within 24 hours, or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limits in accordance with the Surveillance Frequency Control Program.

4.4.4.2 The pressurizer proportional heater groups shall be determined to be OPERABLE.

(a) In accordance with the Surveillance Frequency Control Program by verifying emergency power is available to the heater groups, and

(b) In accordance with the Surveillance Frequency Control Program by verifying that the summed power consumption of the two proportional heater groups is ≥ 150 KW.

Note 1: Action (d) is not applicable when the second group of required pressurizer heaters is intentionally made inoperable.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection instrumentation shall be demonstrated OPERABLE by:

a. Performing a CHANNEL CHECK of the required containment atmosphere radioactivity monitors in accordance with the Surveillance Frequency Control Program.

b. Performing a CHANNEL CHECK of the containment sump level monitor in accordance with the Surveillance Frequency Control Program.

c. Performing a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitors in accordance with the Surveillance Frequency Control Program.

d. Performing a CHANNEL CALIBRATION of the containment sump level monitor in accordance with the Surveillance Frequency Control Program.

e. Performing a CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitors in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakage, except for primary to secondary leakage, shall be demonstrated to be within each of the above limits by:

a. Performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program during steady state operation except when operating in the shutdown cooling mode*.

b. Monitoring the reactor head flange leakoff temperature in accordance with the Surveillance Frequency Control Program.

4.4.6.2.2 Primary to secondary leakage shall be verified to be ≤ 150 gallons per day through any one SG in accordance with the Surveillance Frequency Control Program*.

4.4.6.2.3 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4.6-1 shall be demonstrated OPERABLE by individually verifying leakage to be within its limit:

a. Prior to entering MODE 2 after each refueling outage,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months, and

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

---

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

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

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

ACTION

Note: The provisions of Specification 3.0.4.c are applicable to ACTION a and b.

a. With the DOSE EQUIVALENT I-131 not within limit:

   1. Verify DOSE EQUIVALENT I-131 ≤ 60 μCi/gm once every 4 hours, and

   2. Restore DOSE EQUIVALENT I-131 within limit within 48 hours.

b. With the DOSE EQUIVALENT XE-133 not within limit, restore DOSE EQUIVALENT XE-133 within limit within 48 hours.

c. With the requirements of ACTION a and/or b not met, or with DOSE EQUIVALENT I-131 > 60 μCi/gm, be in at least HOT STANDBY in 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1 Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 3100 μCi/gm in accordance with the Surveillance Frequency Control Program.*

4.4.8.2 Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm:*

   a. in accordance with the Surveillance Frequency Control Program, and

   b. between 2 and 6 hours after THERMAL POWER change of ≥ 15% RATED THERMAL POWER within a 1 hour period.

* Only required to be performed in MODE 1.

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in SAR Table 5.2-12. The results of these examinations shall be used to update Figures 3.4-2A, 3.4-2B and 3.4-2C.

REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one reactor coolant system vent path consisting of at least two valves in series shall be OPERABLE at each of the following locations:

1. Reactor Vessel Head

2. Pressurizer Steam Space (RCS High Point Vents)

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

ACTION:

a. With less than one vent path from each of the locations OPERABLE, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path(s) is maintained closed; restore the inoperable vent path to OPERABLE status within 30 days or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both vent paths 1 and 2 above inoperable, restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor coolant system vent path shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying flow through the reactor coolant vent system vent paths.

4.4.12.1 Verify both sets of LTOP relief valve isolation valves are open in accordance with the Surveillance Frequency Control Program when the LTOP relief valves are being used for overpressure protection.

4.4.12.2 The RCS vent path shall be verified to be open in accordance with the Surveillance Frequency Control Program** when the vent path is being used for overpressure protection.

4.4.12.3 Verify that each SIT is isolated, when required, in accordance with the Surveillance Frequency Control Program.

4.4.12.4 No additional LTOP relief valve Surveillance Requirements other than those required by the INSERVICE TESTING PROGRAM.

** Except when the vent path is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify this valve is open in accordance with the Surveillance Frequency Control Program.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 1413 and 1539 cubic feet (equivalent to an indicated level between 80.1% and 87.9%, respectively),

c. Between 2200 and 3000 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 624 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, due to boron concentration not within limits, restore the boron concentration to within limits within 72 hours, or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 700 psia within the next 12 hours.

b. With one safety injection tank inoperable due to inability to verify level or pressure, restore the SIT to OPERABLE status within 72 hours, or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 700 psia within the next 12 hours.

c. With one safety injection tank inoperable for reasons other than ACTION a or b, restore the SIT to OPERABLE status within 24 hours, or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 700 psia within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

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

2. Verifying that each safety injection tank isolation valve (2CV-5003-1, 2CV-5023-1, 2CV-5043-2, and 2CV-5063-2) is open.

* With pressurizer pressure ≥ 700 psia.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program and within 6 hours after each solution volume increase of ≥ 5% of indicated tank level that is not the result of addition from the RWT, by verifying the boron concentration of the safety injection tank solution.

c. In accordance with the Surveillance Frequency Control Program when the RCS pressure is above 2000 psia, by verifying that power to the isolation valve operator is removed by maintaining the motor circuit breaker open under administrative control.

d. In accordance with the Surveillance Frequency Control Program by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

   1. When the RCS pressure exceeds 700 psia, and

   2. Upon receipt of a safety injection test signal.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the following valves are in the indicated positions with power to the 2CV-5101-1 and 2CV-5102-2 valve operators removed:

Valve Number	Valve Function	Valve Position
2CV-5101-1	HPSI Hot Leg Injection Isolation	Closed
2CV-5102-2	HPSI Hot Leg Injection Isolation	Closed
2BS-26	RWT Return Line	Open

b. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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

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

2. At least once daily of the areas affected within containment if containment has been entered that day, and during the final entry when CONTAINMENT INTEGRITY is established.

d. In accordance with the Surveillance Frequency Control Program by a visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. In accordance with the Surveillance Frequency Control Program, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on SIAS and RAS test signals.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to the INSERVICE TESTING PROGRAM:

   1. High-Pressure Safety Injection pump ≥ 1360.4 psid with 90 °F water.

   2. Low-Pressure Safety Injection pump ≥ 156.25 psid with 90 °F water.

g. In accordance with the Surveillance Frequency Control Program by verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

<u>LPSI System</u>
<u>Valve Number</u>

   a. 2CV-5037-1
   b. 2CV-5017-1
   c. 2CV-5077-2
   d. 2CV-5057-2

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

<u>HPSI System – Single Pump</u>

The sum of the injection line flow rates, excluding the highest flow rate is greater than or equal to 570 gpm.

<u>LPSI System – Single Pump</u>

   a. Injection Leg 1, ≥ 1059 gpm
   b. Injection Leg 2, ≥ 1059 gpm
   c. Injection Leg 3, ≥ 1059 gpm
   d. Injection Leg 4, ≥ 1059 gpm

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. An available borated water volume of between 384,000 and 503,300 gallons

b. Between 2500 and 3000 ppm of boron,

c. A minimum solution temperature of 40 °F, and

d. A maximum solution temperature of 110 °F

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWT shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

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

2. Verifying the boron concentration of the water.

b. In accordance with the Surveillance Frequency Control Program by verifying the RWT temperature.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. In accordance with the Surveillance Frequency Control Program by verifying that each containment isolation manual valve and blind flange (Note 1) that is located outside containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative control as permitted by Specification 3.6.3.1.

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

c. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals in accordance with the Containment Leakage Rate Testing Program.

d. Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days by verifying each containment isolation manual valve and blind flange (Note 1) that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls as permitted by Specification 3.6.3.1.

Note 1: Valves and blind flanges in high radiation areas may be verified by use of administrative means.

SURVEILLANCE REQUIREMENTS

4.6.1.3.1 Each containment air lock shall be demonstrated OPERABLE as specified in the Containment Leakage Rate Testing Program[5,6].

4.6.1.3.2 Each containment air lock interlock shall be demonstrated OPERABLE by testing the air lock interlock mechanism in accordance with the Surveillance Frequency Control Program[7].

---

[5] Leakrate results shall also be evaluated against the acceptance criteria of specification 3.6.1.2.

[6] An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

[7] This surveillance requirement is only required to be performed upon entry or exit through the associated containment air lock.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE AND AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 The combination of containment internal pressure and average air temperature shall be maintained within the region of acceptable operation shown on Figure 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the point defined by the combination of containment internal pressure and average air temperature outside the region of acceptable operation shown on Figure 3.6-1, restore the combination of containment internal pressure and average air temperature to within the above limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure and average air temperature shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program. The containment average air temperature shall be the temperature of the air in the containment HVAC common return air duct upstream of the fan/cooler units.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.6 The containment purge supply and exhaust isolation valves shall be closed and handswitch keys removed.

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

ACTION:

With one or more containment purge supply and/or exhaust isolation valves not closed with the handswitch keys removed, place the valve(s) in the closed position with handswitch keys(s) removed within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The containment purge supply and exhaust isolation valves shall be determined closed in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION, COOLING, AND pH CONTROL SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWT on a Containment Spray Actuation Signal (CSAS) and automatically transferring suction to the containment sump on a Recirculation Actuation Signal (RAS). Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With both containment spray systems inoperable (Note 1):

1. Within 1 hour verify both CREVS trains are OPERABLE, and

2. Restore at least one containment spray system to OPERABLE status within 24 hours.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

1. Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.

2. Verifying that the system piping is full of water from the RWT to at least elevation 505' (equivalent to > 12.5% indicated narrow range level) in the risers within the containment.

b. Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head when tested pursuant to the INSERVICE TESTING PROGRAM.

Note 1: ACTION b is not applicable when the second containment spray system is intentionally made inoperable.

SURVEILLANCE REQUIREMENTS (Continued)

c. In accordance with the Surveillance Frequency Control Program, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on CSAS and RAS test signals.

   2. Verifying that upon a RAS test signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

   3. Verifying that each spray pump starts automatically on a CSAS test signal.

d. Verify each spray nozzle is unobstructed following maintenance which could result in nozzle blockage.

CONTAINMENT SUMP BUFFERING AGENT

LIMITING CONDITION FOR OPERATION

3.6.2.2 The buffering agent baskets shall contain ≥ 308 ft^3 of sodium tetraborate (NaTB) decahydrate.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the buffering agent not within limits, restore the buffering agent to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and be in at least HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The buffering agent shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the buffering agent baskets contain ≥ 308 ft^3 of NaTB decahydrate.

b. In accordance with the Surveillance Frequency Control Program by verifying that a sample from the buffering agent baskets provides adequate pH adjustment of borated water.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment cooling group shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

1. Verifying that service water flow rate to the group of cooling units is ≥ 1250 gpm and that each group has two operable fans.

2. Addition of a biocide to the service water during the surveillance in 4.6.2.3.a.1 above, whenever service water temperature is between 60 °F and 80 °F.

b. In accordance with the Surveillance Frequency Control Program by:

1. Starting (unless already operating) each operational cooling unit from the control room.

2. Verifying that each operational cooling unit operates for at least 15 minutes.

c. In accordance with the Surveillance Frequency Control Program by verifying that each cooling unit starts automatically on a CCAS test signal.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position.

4.6.3.1.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to the INSERVICE TESTING PROGRAM.

4.6.3.1.4 The containment purge supply and exhaust isolation valves shall be demonstrated OPERABLE as specified in the Containment Leakage Rate Testing Program.

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two emergency feedwater pumps and associated flow paths shall be OPERABLE with:

a. One motor driven pump capable of being powered from an OPERABLE emergency bus, and

b. One turbine driven pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

NOTE: Specification 3.0.4.b is not applicable.

With one emergency feedwater pump inoperable, restore the inoperable pump to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater pump shall be demonstrated OPERABLE:

a In accordance with the Surveillance Frequency Control Program by:

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

b. In accordance with the INSERVICE TESTING PROGRAM by:

1. Verifying the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head. This surveillance requirement is not required to be performed for the turbine driven EFW pump until 24 hours after exceeding 700 psia in the steam generators.

SURVEILLANCE REQUIREMENTS (Continued)

c In accordance with the Surveillance Frequency Control Program by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on MSIS or EFAS test signals.

2. Verifying that the motor driven pump starts automatically upon receipt of an EFAS test signal.

3. Verifying that the turbine driven pump steam supply MOV opens automatically upon receipt of an EFAS test signal.

d. By verifying proper alignment of the required EFW flow paths by verifying flow from the condensate storage tank to each steam generator. This SR is required to be verified prior to entering MODE 2 whenever plant has been in MODES 4, 5, 6, or defueled for > 30 days.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 At least one condensate storage tank (CST) shall be OPERABLE with a minimum contained water volume of either:

a. 160,000 gallons in either 2T41A or 2T41B, or

b. A minimum of 267,000 gallons of water is available in condensate storage tank, T41B, when required for both units. A minimum of 160,000 gallons of water is available in T41B when only required for Unit 2.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the required condensate storage tank inoperable, within 4 hours either:

a. Restore at least one CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the service water system as a backup supply to the emergency feedwater pumps and restore at least one condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The above required condensate storage tank shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the contained water volume is within its limits when the tank is the supply source for the emergency feedwater pumps.

4.7.1.3.2 The service water system shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that at least one service water loop is operating and that the service water system - emergency feedwater system isolation valves are either open or OPERABLE whenever the service water system is the supply source for the emergency feedwater pumps.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	In accordance with the Surveillance Frequency Control Program
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) In accordance with the Surveillance Frequency Control Program, whenever the gross activity determination is greater than 10% of the allowable iodine limit. b) In accordance with the Surveillance Frequency Control Program, whenever the gross activity determination is below 10% of the allowable iodine limit.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 90 °F when the pressure of either coolant in the steam generator is > 275 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to ≤ 275 psig within 30 minutes, and

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

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generators shall be determined to be < 275 psig in accordance with the Surveillance Frequency Control Program when the temperature of either the primary or secondary coolant is < 90 °F.

PLANT SYSTEMS

3/4.7.3 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

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

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on CCAS, MSIS and RAS test signals.

3/4.7.4 EMERGENCY COOLING POND

LIMITING CONDITION FOR OPERATION

3.7.4.1 The emergency cooling pond (ECP) shall be OPERABLE with:

a. A minimum contained water volume of 70 acre-feet.

b. An average water temperature of ≤ 100 °F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the volume and/or temperature requirements of the above specification not satisfied or, with the requirements of Action b not met, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. If degradation is noted pursuant to 4.7.4.1.d below or by other inspection, perform an evaluation to determine that the ECP remains acceptable for continued operation within 7 days.

SURVEILLANCE REQUIREMENTS

4.7.4.1 The ECP shall be determined OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the indicated water level of the ECP is greater than or equal to that required for an ECP volume of 70 acre-feet.

b. In accordance with the Surveillance Frequency Control Program during the period of June 1 through September 30 by verifying that the pond's average water temperature at the point of discharge from the pond is within its limit.

c. In accordance with the Surveillance Frequency Control Program by making soundings of the pond and verifying:

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.

d. In accordance with the Surveillance Frequency Control Program by performance of a visual inspection of the ECP to verify conformance with design requirements.

3/4.7.5 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.5.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Dardanelle Reservoir exceeds 350 feet Mean Sea Level USGS datum, at the intake structure.

APPLICABILITY: When a flood warning exists at the facility site.

ACTION:

With the water level at the intake structure above elevation 350 feet Mean Sea Level USGS datum, initiate and complete within 4 hours, closure of the openings and penetrations listed in Table 3.7-6 using the equipment listed in Table 3.7-6.

SURVEILLANCE REQUIREMENTS

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

a. Measurement in accordance with the Surveillance Frequency Control Program when the water level is below elevation 350 feet Mean Sea Level USGS datum, and

b. Measurement in accordance with the Surveillance Frequency Control Program when the water level is equal to or above elevation 350 feet Mean Sea Level USGS datum.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 Each control room emergency air conditioning system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

1. Starting each unit from the control room, and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤ 84 °F D.B.

b. In accordance with the Surveillance Frequency Control Program by verifying a system flow rate of 9900 cfm ± 10%.

4.7.6.1.2 Each control room emergency air filtration system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the system operates for at least 15 minutes.

b. In accordance with the Surveillance Frequency Control Program by verifying that on a control room high radiation signal, either actual or simulated, the system automatically isolates the control room and switches into a recirculation mode of operation.

c. By performing the required Control Room Emergency Ventilation filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

d. Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

Next Page is 3/4 7-27

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

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

1. Either decontaminated and repaired, or

2. Disposed of in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

4.7.9.1.2 Test Frequencies – Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequencies described below.

a. Sources in use – In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive material:

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE: (Note 1)

a. In accordance with the Surveillance Frequency Control Program by:

1. Verifying the fuel level in the day fuel tank.

2. deleted

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts from a standby condition and accelerates to at least 900 rpm in ≤ 15 seconds. (Note 2)

5. Verifying the generator is synchronized, loaded to an indicated 2600 to 2850 Kw and operates for ≥ 60 minutes. (Notes 3 & 4)

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

b. deleted

Note 1
All planned diesel generator starts for the purposes of these surveillances may be preceded by prelube procedures.

Note 2
This diesel generator start from a standby condition in ≤ 15 sec. shall be accomplished at least once every 184 days. All other diesel generator starts for this surveillance may be in accordance with vendor recommendations.

Note 3
Diesel generator loading may be accomplished in accordance with vendor recommendations such as gradual loading.

Note 4
Momentary transients outside this load band due to changing loads will not invalidate the test. Load ranges are allowed to preclude over- loading the diesel generators.

c. In accordance with the Surveillance Frequency Control Program by:

1. Deleted

2. Verifying during shutdown that the automatic sequence time delay relays are OPERABLE at their setpoint ± 10% of the elapsed time for each load block.

3. Verifying during shutdown the generator capability to reject a load of greater than or equal to its associated single largest post-accident load, and maintain voltage at 4160 ± 500 volts and frequency at 60 ± 3 Hz.

4. Verifying during shutdown the generator capability to reject a load of 2850 Kw without exceeding 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal, whichever is lower.

5. Simulating during shutdown a loss of offsite power by itself, and:

   a. Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b. Verifying the diesel starts from a standby condition on the undervoltage auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected shutdown loads through the time delay relays and operates for ≥ 5 minutes while its generator is loaded with the shutdown loads.

6. Verifying during shutdown that on a Safety Injection Actuation Signal (SIAS) actuation test signal (without loss of offsite power) the diesel generator starts on the auto-start signal and operates on standby for ≥ 5 minutes.

11. Verifying during shutdown the diesel generator's capability to:

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

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

    c) Proceed through its shutdown sequence.

12. Verifying during shutdown that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the auto-connected emergency (accident) loads with offsite power.

13. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

d. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 900 rpm in ≤ 15 seconds.

LIMITING CONDITION FOR OPERATION

3.8.1.3 The stored diesel fuel oil shall be within limits for each required diesel generator.

APPLICABILITY: When associated diesel generator is required to be OPERABLE.

ACTION:

With the volume of the stored diesel fuel oil less than 22,500 gallons for either fuel oil storage tank or the new or stored fuel oil properties outside the limits of the Diesel Fuel Oil Testing Program, perform the following as appropriate: (Note – Separate ACTION entry is allowed for each diesel generator.)

1. If one or more fuel storage tanks contain less than 22,500 gallons and greater than 17,446 gallons, restore the fuel oil volume to within limits within 48 hours.

2. If the stored fuel oil total particulates are not within limits for one or more diesel generators, restore fuel oil total particulates to within limits within 7 days.

3. If new fuel oil properties are not within limits for the one or more diesel generators, restore stored fuel oil properties to within limits within 30 days.

4. If ACTION 1 is not met within the allowable outage time or is outside the allowable limits, or if ACTION 2 or 3 is not met within the allowable outage time, then immediately declare the associated diesel generator inoperable.

SURVEILLANCE REQUIREMENTS

4.8.1.3.1 In accordance with the Surveillance Frequency Control Program verify the fuel oil storage tank contains ≥ 22,500 gallons of fuel.

4.8.1.3.2 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.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION – OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized with tie breakers open between redundant busses:

4160 volt Emergency Bus # 2A3

4160 volt Emergency Bus # 2A4

480 volt Emergency Bus # 2B5

480 volt Emergency Bus # 2B6

120 volt A.C. Vital Bus # 2RS1

120 volt A.C. Vital Bus # 2RS2

120 volt A.C. Vital Bus # 2RS3

120 volt A.C. Vital Bus # 2RS4

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE with tie breakers open between redundant busses in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

ELECTRICAL POWER SYSTEMS

A.C. DISTRIBUTION – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses shall be OPERABLE:

1 - 4160 volt Emergency Bus

1 - 480 volt Emergency Load Center Bus

4 - 480 volt Motor Control Center Busses

2 - 120 volt A.C. Vital Busses

APPLICABILITY: MODES 5 and 6

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, immediately suspend core alterations, the movement of irradiated fuel assemblies, and any operations involving positive reactivity additions.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

DC SOURCES – OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required full capacity chargers inoperable:

   i. Restore the battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours, and

   ii. Verify battery float current ≤ 2 amps once per 12 hours.

b. With one DC electrical power subsystem inoperable for reasons other than ACTION ‘a’ above, restore the inoperable DC electrical power subsystem to OPERABLE status within 2 hours.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 In accordance with the Surveillance Frequency Control Program by verifying that the battery terminal voltage is greater than or equal to the minimum established float voltage.

4.8.2.3.2 In accordance with the Surveillance Frequency Control Program by verifying that each battery charger supplies ≥ 300 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours or, by verifying that 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.

4.8.2.3.3 In accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is adequate to supply, and maintain in OPERABLE status, required emergency loads for the design duty cycle when subjected to a battery service test. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this Surveillance. The battery performance discharge test required by Surveillance Requirement 4.8.3.6 may be performed in lieu of the battery service test once per 60 months.

DC SOURCES – SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

1 - 125-volt DC bus, and

1 - 125-volt battery bank and charger supplying the above DC bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the required battery charger inoperable:

   i. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours, and

   ii. Verify battery float current ≤ 2 amps once per 12 hours.

b. With the requirements of ACTION ‘a’ not met or with the above complement of DC equipment and bus otherwise inoperable, immediately suspend core alterations, the movement of irradiated fuel assemblies, and any operations involving positive reactivity additions.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus shall be determined OPERABLE and energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirements 4.8.2.3.1, 4.8.2.3.2, and 4.8.2.3.3; however, while each of these Surveillance Requirements must be met, Surveillance Requirements 4.8.2.3.2 and 4.8.2.3.3 are not required to be performed.

SURVEILLANCE REQUIREMENTS

4.8.3.1 In accordance with the Surveillance Frequency Control Program by verifying that each battery float current is ≤ 2 amps. This Surveillance is not required when battery terminal voltage is less than the minimum established float voltage of Surveillance Requirement 4.8.2.3.1.

4.8.3.2 In accordance with the Surveillance Frequency Control Program by verifying that each battery pilot cell float voltage is ≥ 2.07 V.

4.8.3.3 In accordance with the Surveillance Frequency Control Program by verifying that each battery connected cell electrolyte level is greater than or equal to minimum established design limits.

4.8.3.4 In accordance with the Surveillance Frequency Control Program by verifying that each battery pilot cell temperature is greater than or equal to minimum established design limits.

4.8.3.5 In accordance with the Surveillance Frequency Control Program by verifying that each battery connected cell float voltage is ≥ 2.07 V.

4.8.3.6 In accordance with the Surveillance Frequency Control Program by verifying the battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this Surveillance. In addition, the performance discharge test shall be performed:

a. At least once per 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating, and

b. At least once per 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating.

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of the reactor coolant and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of ≥ 2500 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at ≥ 40 gpm of ≥ 2500 ppm boric acid solution until K_{eff} is reduced to ≤ 0.95 or the boron concentration is restored to ≥ 2500 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant and the refueling canal shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program.

---

* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable, determine the boron concentration of the reactor coolant system at least once per 12 hours.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. A CHANNEL CHECK in accordance with the Surveillance Frequency Control Program,

b. A CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program, and

c. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS.

CONTAINMENT BUILDING PENETRATION

LIMITING CONDITION FOR OPERATION

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

a. The equipment door is capable* of being closed,

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

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

1. Closed* by a manual or automatic isolation valve, blind flange, or equivalent, or

2. Capable* of being closed by an OPERABLE containment purge and exhaust isolation system.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment penetrations shall be determined to be in its above required conditions within 72 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS or movement of irradiated fuel in the containment.

---

* Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. Administrative controls shall ensure that appropriate personnel are aware that when containment penetrations, including both personnel airlock doors and/or the equipment door are open, a specific individual(s) is designated and available to close the penetration following a required evacuation of containment, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door and/or the equipment door be capable of being quickly removed.

Next page is 3/4 9-6

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS.

CRANE TRAVEL – SPENT FUEL POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pool.

APPLICABILITY: With fuel assemblies in the spent fuel pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 The crane electrical power disconnect which prevents crane travel over the spent fuel pool shall be verified open under administrative control in accordance with the Surveillance Frequency Control Program, or the crane travel interlock which prevents crane travel over the spent fuel pool shall be demonstrated OPERABLE within 4 hours prior to each use of the crane for lifting loads in excess of 2000 pounds.

REFUELING OPERATIONS

SHUTDOWN COOLING AND COOLANT CIRCULATION

SHUTDOWN COOLING – ONE LOOP

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

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

b. The shutdown cooling loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 A shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of ≥ 2000 gpm in accordance with the Surveillance Frequency Control Program.

WATER LEVEL – REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel while in MODE 6, except during latching and unlatching of CEAs.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.9 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and in accordance with the Surveillance Frequency Control Program thereafter during movement of fuel assemblies or CEAs.

SPENT FUEL POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.10 The water level in the spent fuel pool shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program when irradiated fuel assemblies are in the spent fuel pool.

LIMITING CONDITION FOR OPERATION

3.9.12.a Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.95 w/o U-235. The provisions of Specification 3.0.3 are not applicable.

3.9.12.b Storage in the spent fuel pool shall be further restricted by the limits specified in Table 3.9-1. The provisions of Specification 3.0.3 are not applicable.

3.9.12.c The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 2000 parts per million.

APPLICABILITY: During storage of fuel in the spent fuel pool

ACTION:

Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined a fuel assembly has been placed in an incorrect location until such time as the correct storage location is determined. Move the assembly to its correct location before resumption of any other fuel movement.

Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined the pool boron concentration is less than 2001 ppm, until such time as the boron concentration is increased to 2001 ppm or greater.

SURVEILLANCE REQUIREMENTS

4.9.12.a Verify all fuel assemblies to be placed in the spent fuel pool have an initial enrichment of less than or equal to 4.95 w/o U-235 by checking the assemblies' design documentation.

4.9.12.b Verify all fuel assemblies to be placed in the spent fuel pool are within the limits of Table 3.9-1 by checking the assemblies' design and burnup documentation.

4.9.12.c Verify in accordance with the Surveillance Frequency Control Program the spent fuel pool boron concentration is greater than 2000 ppm.

4.9.12.d Verify Metamic properties are in accordance with, and are maintained within the limits of, the Metamic Coupon Sampling Program.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

a. With any CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each CEA required either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of Functional Unit 14 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The linear heat rate limit shall be maintained by either:

   1. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

   2. Operating within the region of acceptable operation as specified in the CORE OPERATING LIMITS REPORT using any operable CPC channel (when COLSS is out of service.)

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With any of the above limits being exceeded while any of the above requirements are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of the above Specification, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS in which any of the above requirements are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within its limits during PHYSICS TESTS above 5% of RATED THERMAL POWER in which any of the above requirements are suspended.

SPECIAL TEST EXCEPTIONS

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

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

b. The reactor trip setpoints of the OPERABLE power level channels are set at ≤ 20% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during startup and PHYSICS TESTS.

4.10.3.2 Each wide range logarithmic and power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the incore detection system during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.5 The minimum temperature for criticality limits of Specification 3.1.1.5 may be suspended during low temperature PHYSICS TESTS provided:

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

b. The reactor trip setpoints on the OPERABLE Linear Power Level - High neutron flux monitoring channels are set at ≤ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation shown on Figure 3.4-2.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

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

b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figure 3.4-2, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the engineering evaluation required by Specification 3.4.9.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The Reactor Coolant System temperature and pressure relationship shall be verified to be within the acceptable region for operation of Figure 3.4-2 in accordance with the Surveillance Frequency Control Program.

4.10.5.2 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program.

4.10.5.3 Each Logarithmic Power Level and Linear Power Level channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1 The quantity of radioactive material contained in each unprotected outside temporary radioactive liquid storage tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material exceeding the above limit, immediately suspend all additions of radioactive material to the affected tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluents Release Report pursuant to Specification 6.9.3.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1 The quantity of radioactive material contained in each unprotected outside temporary radioactive liquid storage tank shall be determined to be within the above limit by analyzing a representative sample of the contents of the tank in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

---

* Tanks included in this specification are those outdoor temporary tanks that 1) are not surrounded by liners, dikes, or walls capable of holding the tank contents, and 2) do not have overflows and surrounding area drains connected to the liquid radwaste treatment system.

3/4.11.2 GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 82,400 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to the condition in the next Radioactive Effluent Release Report pursuant to Specification 6.9.3.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank and the reactor coolant activity exceeds the limits of Specification 3.4.8.

6.5 PROGRAMS AND MANUALS

6.5.2 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:

c. Preventive maintenance and periodic visual inspection requirements; and

d. Integrated leak test requirements for each system at a Frequency in accordance with the Surveillance Frequency Control Program. The provisions of Surveillance Requirements 4.0.2 are applicable.

6.5.3 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.

6.5.4 Radioactive Effluent Controls Program

This program conforms with 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 to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS, conforming to 10 CFR 50, Appendix I;

6.5.12 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 inleakage 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 assessment of the CRE boundary.

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

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

6.5.13 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 4.0.2 and SR 4.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance frequencies.

6.5.14 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 require 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.

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

d. Proposed changes that do not meet the criteria of 6.5.14b above 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).

6.5.17 Metamic Coupon Sampling Program

A coupon surveillance program will be implemented to maintain surveillance of the Metamic absorber material under the radiation, chemical, and thermal environment of the SFP. The purpose of the program is to establish the following:

- Coupons will be examined on a two year basis for the first three intervals with the first coupon retrieved for inspection being on or before October 31, 2009 and thereafter at increasing intervals over the service life of the inserts.
- Measurements to be performed at each inspection will be as follows:
  a. Physical observations of the surface appearance to detect pitting, swelling or other degradation,
  b. Length, width, and thickness measurements to monitor for bulging and swelling
  c. Weight and density to monitor for material loss, and
  d. Neutron attenuation to confirm the B-10 concentration or destructive chemical testing to determine the boron content.
- The provisions of SR 4.0.2 are applicable to the Metamic Coupon Sampling Program.
- The provisions of SR 4.0.3 are not applicable to the Metamic Coupon Sampling Program.

6.5.18 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.

ATTACHMENT 5 to

2CAN021802

PROPOSED CHANGES TO TECHNICAL SPECIFICATION BASES PAGES
(Information Only)

SR 4.0.1 (continued)

Some examples of this process are:

a. Emergency Feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 700 psi. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

b. High Pressure Safety Injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 4.0.2

SR 4.0.2 establishes the requirements for meeting the specified Frequency interval for Surveillances and any ACTION with an AOT that requires the periodic performance of the ACTION on a "once per…" interval.~~ For those SR intervals established on a STAGGERED TEST BASIS, the 25% extension is applied to the stated frequency divided by the number of trains/channels associated with the system. For example, given an SR on a four channel system required to be performed TA (every 123 days) on a STAGGERED TEST BASIS, the 25% extension is applied on a per-channel basis (123/4 x 1.25 = 38 days, 10½ hours).~~

SR 4.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities). It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval.

When a Section 6.5, "Programs and Manuals," specification states that the provisions of SR 4.0.2 are applicable, a 25% extension of the testing interval, whether stated in the specification or incorporated by reference, is permitted.

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 4.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 4.0.2 does not apply is in the Containment Leakage Rate Testing Program required by 10 CFR 50, Appendix J, and the inservice testing of pumps and valves in accordance with applicable American Society of Mechanical Engineers Operation and Maintenance Code, as required by 10 CFR 50.55a.. These programs establish testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot, in and of themselves, extend a test interval specified in the regulations directly or by reference.

The ACTION statements applicable to a trippable but misaligned or inoperable CEA include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements brings the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available SHUTDOWN MARGIN which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified ~~on a nominal basis of once per 12 hours~~in accordance with the Surveillance Frequency Control Program with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The average CEA drop time restriction is consistent with the assumed CEA drop time used in the accident analysis. The maximum CEA drop time restriction is used to limit the CEA drop time distribution about the average to that used in the accident analysis. Measurement with T_{avg} $\geq$ 525 °F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

The establishment of LSSS and LCOs require that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base load, load following, etc.) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors are defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. The CHANNEL FUNCTIONAL TEST frequency in controlled under the Surveillance Frequency Control Program~~of once per 123 days is to be performed on a STAGGERED TEST BASIS~~.

The RPS Matrix Logic channels and the Initiation Logic channels are listed as separate functional units in Table 3.3-1 and are grouped together in the corresponding surveillance Table 4.3-1 as a single functional unit listed as Reactor Protection System (RPS) Logic. The RPS Logic contains six Matrix Logic channels and four Initiation Logic channels. ~~For surveillance testing purposes, the RPS Logic is considered to have four channels or n = 4 with respect to STAGGERED TEST BASIS.~~ The associated CHANNEL FUNCTIONAL TEST requirements are performed during the individual channel PPS test. The six RPS Matrix Logic channels are divided up for testing purposes as follows: Matrix AB is tested with channel A, matrices BC and BD are tested with channel B, matrices AC and CD are tested with channel C, and matrix AD is tested with channel D.~~ This testing methodology is supported by the analysis that was performed to extend the surveillance interval to the once per 123 day frequency and also satisfies the STAGGERED TEST BASIS requirements for the RPS Matrix Logic channels.~~

Table 4.3-1 requires verification (in accordance with the Surveillance Frequency Control Program~~once per 12 hours~~) that CPC indicated flow rate is less than or equal to the RCS total flow rate measured using either reactor coolant pump differential pressure instrumentation or calorimetric calculations (see Note 7). This calibration requirement ensures that the CPC calculation of DNBR uses a conservative value of RCS total flow rate. The calibration check is typically performed by comparing CPC and reactor coolant pump differential pressure based COLSS relative mass flows (in terms of the fraction of design flow rate). When COLSS is out of service, the calibration of CPC flow is performed by comparing to a calorimetric calculation of the flow rate. Appropriate flow measurement uncertainties for either method of determining the actual flow rate are included in the determination of CPC DNBR uncertainty addressable constant BERR1 (using methodology described in CEN-356(V)-P-A). The flow measurement uncertainty accounts for process and instrumentation uncertainties as well as uncertainties associated with calibration of the COLSS flow measurement algorithm based on pump casing curves, validated calorimetric flow measurements and detailed simulations of RCS flow.

Table 4.3-2 requires the Automatic Actuation Logic channels for each of the associated ESFAS functional units to have a CHANNEL FUNCTIONAL TEST performed in accordance with the Surveillance Frequency Control Program~~on a once per 123 day frequency on a STAGGERED TEST BASIS~~. These testing requirements also apply to the six ESFAS Matrix Logic channels and the four ESFAS Initiation Logic channels. ~~For surveillance testing purposes, the ESFAS Matrix Logic channels and the ESFAS Initiation Logic channels are considered to have four channels or n = 4 with respect to STAGGERED TEST BASIS.~~ The ESFAS Matrix Logic channels are divided up for testing purposes like the RPS Matrix Logic channels. ~~This testing methodology is supported by the analysis that was performed to extend the surveillance interval to the once per 123 day frequency and also satisfies the STAGGERED TEST BASIS requirements for the ESFAS Matrix Logic channels.~~

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses.

No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements, provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times or 3) utilizing allocated response time for selected sensors. Topical Report CE NPSD-1167-A, “Elimination of Pressure Sensor Response Time Testing Requirements,” provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the Topical Report. Response time verification for sensor types must be demonstrated by test. The allocation of sensor response times must be verified prior to placing a new component in operation and re-verified after maintenance that may adversely affect the sensor response time.

Plant Protective System (PPS) logic is designed for operation as a 2-out-of-3 logic, although normally it is operated in a 2-out-of-4 mode.

The RPS Logic consists of everything downstream of the bistable relays and upstream of the Reactor Trip Circuit Breakers. The RPS Logic is divided into two parts, Matrix Logic, and Initiation Logic. Failures of individual bistables and their relays are considered measurement channel failures.

The ESFAS Logic consists of everything downstream of the bistable relays and upstream of the subgroup relays. The ESFAS Logic is divided into three parts, Matrix Logic, Initiation Logic, and Actuation Logic. Failures of individual bistables and their relays are considered measurement channel failures.

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting matrix wiring between bistable relay cards, up to, but not including the matrix relays. Matrix contacts on the bistable relay cards are excluded from the Matrix Logic definition since they are addressed as part of the measurement channel.

3/4.4.6.2 REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The Surveillance Requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The limit of 150 gallons per day through any one SG is based on the operational leakage performance criterion in NEI 97-06, *Steam Generator Program Guidelines* which states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day". The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

The 150 gallons per day limit is measured at room temperature as described in EPRI, *Pressurized Water Reactor Primary-to-Secondary Leak Guidelines*. 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.

For 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 ~~of 72 hours~~ is a reasonable interval to trend primary to secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI *Pressurized Water Reactor Primary-to-Secondary Leak Guidelines*.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.8 SPECIFIC ACTIVITY (continued)

ACTION a (continued)

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The AOT of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a MSLB or SGTR occurring during this time period.

ACTION b

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The AOT of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a MSLB or SGTR occurring during this time period.

ACTION c

If the required action and associated AOT of ACTION a and/or b is not met, or if the DOSE EQUIVALENT I-131 is > 60.0 μCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within the following 30 hours. The AOTs are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

A Note modifies the Surveillance Requirements (SRs) to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the surveillance to be performed in those MODES, prior to entering MODE 1.

SR 4.4.8.1

SR 4.4.8.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant in accordance with the Surveillance Frequency Control Program~~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. The degassed activities may be accounted for by use of a solubility correction factor for gases that remain in solution. 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 Surveillance~~7-day~~ Frequency considers the low probability of a gross fuel failure during this time.

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 4.4.8.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

3/4.4.8 SPECIFIC ACTIVITY (continued)

SR 4.4.8.2

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~~14-day~~ Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored in accordance with the Surveillance Frequency Control Program~~every 7 days~~.

The Frequency, between 2 and 6 hours after a power change > 15% RTP within a 1- hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

REFERENCES

1. 10 CFR 50.67.

2. 10 CFR 50.36.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2.1.5 of the SAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates do not exceed the design assumptions and satisfy the stress limits for cyclic operation.

As used in this specification, the term ‘Reactor Coolant Temperature T_c’ is that indication which is representative of the fluid temperature entering the reactor vessel. With one or more RCPs operating, the T_c indicator(s) associated with the operating RCP(s) should be used to verify compliance with pressure/temperature (P/T) limits. When more than one RCP is in operation or during natural circulation operations, the loop with the lowest T_c temperature should be used. With Shutdown Cooling (SDC) in service and no RCPs in operation, the SDC return temperature should be used to verify compliance with P/T limits. The specific indicator(s) used to verify compliance with P/T limitations are based on engineering determinations and delineated in implementing procedures.

Operation within the limits of the appropriate heatup and cooldown curves assure the integrity of the reactor vessel against fracture induced by combined thermal and pressure stresses. As the vessel is subjected to increasing fluence, the toughness of the limiting material continues to decline, and even more restrictive pressure/temperature limits must be observed.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 50.67 during accident conditions.

Surveillance Requirement (SR) 4.6.1.1.a requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those containment isolation valves outside containment and capable of being mis-positioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the ~~31-day~~Surveillance Frequency, controlled under the Surveillance Frequency Control Program, is based on engineering judgment and was chosen to provide added assurance of the correct positions. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

SR 4.6.1.1.d requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. Containment isolation valves that are open under administrative controls are not required to meet the SR during the time that they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note associated with SR 4.6.1.1.a and SR 4.6.1.1.d applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, 4 and for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak design basis loss of coolant accident pressure, P_a, of 58 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ during the performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM (continued)

The CSS and the Containment Cooling System (CCS) provide post-accident cooling and mixing of the containment atmosphere; however, the CCS is not redundant to the CSS. The CSS also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

If one inoperable CSS train cannot be restored to an OPERABLE status within the allowable outage time (AOT), the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within the following 6 hours. Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (reference CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state.

With two required CSS trains inoperable, at least one of the required CSS trains must be restored to OPERABLE status within 24 hours. Both trains of CCS must be OPERABLE (refer to LCO 3.6.2.3) and both CREVS trains must be verified to be OPERABLE within 1 hour (refer to LCO 3.7.6.1). ACTION b is modified by a Note stating it is not applicable if the second CSS train is intentionally declared inoperable. The ACTION does not apply to voluntary removal of redundant systems or components from service. The ACTION is only applicable if one train is inoperable for any reason and the second train is discovered to be inoperable, or if both trains are discovered to be inoperable at the same time. The components in this degraded condition are capable of providing greater than 100% of the heat removal needs after an accident. The AOT is based on WCAP-16125-NP-A, “Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown,” Revision 2, August 2010, which demonstrated that the 24-hour AOT is acceptable based on the redundant heat removal capabilities afforded by the CCS, the iodine removal capability of the CREVS, the infrequent use of the ACTION, and the small incremental effect on plant risk. If at least one CSS train is not restored to OPERABLE status within 24 hours, the plant must be shutdown as described above.

SR 4.6.2.1.a.2 requires verification of the CSS header level in accordance with the Surveillance Frequency Control Program~~once every 31 days~~. Verifying that the CSS header piping is full of water to the Elevation 505’ level minimizes the time required to fill the header upon a Containment Spray Actuation Signal (CSAS). This ensures that spray flow will be admitted to the Containment Building atmosphere within the time frame assumed in the accident analysis. This SR is not associated with subject matter related to gas accumulation. TS-required systems must always be maintained sufficiently full of water to ensure the specified safety function will be performed when called upon.

3/4.6.2.2 CONTAINMENT SUMP BUFFERING AGENT (continued)

A hydrated form of buffering agent is used because of the high humidity in the containment building during normal operation. Since the buffering agent is hydrated, it is less likely to absorb large amounts of water from the humid atmosphere and will undergo less physical and chemical change than the anhydrous form of buffering agent.

The LOCA radiological consequences analysis takes credit for iodine retention in the sump solution based on the recirculation water pH being ≥ 7.0. The radionuclide releases from the containment atmosphere and the consequences of a LOCA would be increased if the pH of the recirculation water were not adjusted to 7.0 or above.

The required amount of buffering agent is based upon the extreme cases of water volume and pH possible in the containment sump after a large break LOCA. The minimum required volume is the volume of buffering agent that will achieve a sump solution pH of ≥ 7.0 when taking into consideration the calculated sump water volume and boron concentration resulting in the minimum possible pH. The amount of buffering agent needed in the containment building is based on the mass of buffering agent required to achieve the desired pH. However, a required volume is specified, rather than mass, since it is not feasible to weigh the entire amount of buffering agent in containment. The minimum required volume is based on the manufactured density of buffering agent. Since buffering agent can have a tendency to agglomerate from high humidity in the containment building, the density may increase and the volume decrease during normal plant operation. Due to possible agglomeration and increase in density, estimating the minimum volume of buffering agent in containment is conservative with respect to achieving a minimum required pH.

Sufficient buffering agent is required to be available in MODES 1, 2, and 3, because the RCS is at elevated temperature and pressure, providing an energy potential for a LOCA. The potential for a LOCA results in a need for the ability to control the pH of the recirculated coolant.

If it is discovered that the buffering agent in the containment building is not within limits, action must be taken to restore the buffering agent to within limits. During plant operation the containment sump is not accessible and corrections may not be possible. 72 hours is allowed for restoring the buffering agent within limits, where possible, because 72 hours is the same time allowed for restoration of other ECCS components. If the buffering agent cannot be restored within limits within 72 hours, the plant must be brought to a MODE in which the LCO does not apply. The specified Allowed Outage Times for reaching HOT STANDBY and HOT SHUTDOWN were chosen to allow reaching the specified conditions from full power in an orderly manner and without challenging plant systems.

The SR 4.6.2.2.a periodic determination of the volume of buffering agent in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the buffering agent during normal operation. ~~A~~The Surveillance Frequency ~~of 18 months~~is controlled under the Surveillance Frequency Control Program and is required to determine visually that combined a minimum of 308 cubic feet is contained in the buffering agent baskets. This requirement ensures that there is an adequate volume of buffering agent to adjust the pH of the post LOCA sump solution to a value ≥ 7.0.

3/4.6.2.2 CONTAINMENT SUMP BUFFERING AGENT (continued)

The periodic verification is required in accordance with the Surveillance Frequency Control Program~~every 18 months~~, since access to the buffering agent baskets is only feasible during outages, and normal fuel cycles are scheduled for 18 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of buffering agent placed in the containment building.

The SR 4.6.2.2.b requirement to dissolve a representative sample of buffering agent in a sample of borated water provides assurance that the stored buffering agent will dissolve in borated water at the postulated post-LOCA temperatures. Testing must be performed to ensure the solubility and buffering ability of the buffering agent after exposure to the containment environment. A representative sample of 3.07 $\pm$ 0.05 grams of buffering agent from one of the baskets in containment is submerged in 1.0 $\pm$ 0.01 liter of water at a boron concentration of 3248 $\pm$ 30 ppm and at a temperature of 120 $\pm$ 5 °F. The solution is allowed to stand for 4 hours without agitation. The liquid is then decanted from the solution and mixed, the temperature adjusted to 77 $\pm$ 2 °F and the pH measured. At this point, the pH must be $\geq$ 7.0. The representative sample weight is based on the minimum required buffering agent weight of 6804 kilograms (15,000 lbs), which at a manufactured minimum density of 48.7 lbm/ft^3 corresponds to the minimum volume of 308 cubic ft, and assumed post LOCA borated water mass in the sump of approximately 4,644,000 lbm normalized to buffer a 1.0 liter sample. The boron concentration of the test water is representative of the maximum possible boron concentration corresponding to the calculated post LOCA sump volume producing the lowest pH. Agitation of the test solution is prohibited, since an adequate standard for the agitation intensity cannot be specified. The test time of 4 hours is necessary to allow time for the dissolved buffering agent to naturally diffuse through the sample solution. In the post LOCA containment sump, rapid mixing would occur, significantly decreasing the actual amount of time before the required pH is achieved. This would ensure compliance with the Standard Review Plan requirement of a pH $\geq$ 7.0 by the onset of recirculation after a LOCA.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

The containment spray system is redundant to the containment cooling system in providing post-accident cooling and mixing of the containment atmosphere; however, the containment cooling system is not redundant to the containment spray system. As a result of the redundancy of the containment spray system with the containment cooling system, the allowable out-of-service time requirements for the containment cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the containment spray system have been maintained consistent with that assigned other inoperable ESF equipment since the containment spray system also provides a mechanism for removing Iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

Containment penetration isolation response times are not applicable when these administrative controls are applied. When opening a penetration using the allowance of the LCO Note, LCO entry is delayed during the time period the Note is being applied. It is also acceptable to apply these administrative controls for a valve closed in accordance with ACTION b or c. In addition, in accordance with LCO 3.0.4, a penetration in which this Note is being applied may remain open during MODE changes because TS 3.6.3.1 permits continued plant operation in this configuration, provided the valve is closed when the required administrative controls are withdrawn.

With one or more CIVs inoperable in one or more penetrations, the method of penetration isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Examples of isolation barriers that meet this criterion are a closed and de-activated automatic reactor building isolation valve, a closed manual valve, a blind flange, and a check valve (inside containment) with flow through the valve secured. Note that if both CIVs in a given penetration are inoperable while the penetration remains open, LCO 3.0.3 is also applicable until the penetration is isolated by at least one isolation barrier as described above, except as permitted under the aforementioned LCO Note.

With Actions “a”, “b”, or “c” not met (as applicable), the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within the following 6 hours. Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (reference CE NPSD-1186-A, Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG PWRs, October, 2001). In MODE 4 there are more accident mitigation systems available and there is more redundancy and diversity in core heat removal mechanisms than in MODE 5. However, voluntary entry into MODE 5 may be made as it is also an acceptable low-risk state. These Actions are 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.

Unless there is reason to believe the seating capability of the affected valve(s) has degraded, no verification of leakage through the penetration is required. If seat degradation is suspected, a vent or drain within the penetration boundary may be used to verify sufficient seating has taken place. Any noted leak-by should be evaluated in accordance with the Containment Leakage Rate Testing Program of Specification 6.5.16. CIVs closed due to inoperabilities in the respective penetration must be verified to remain in the isolated position ~~once every 31 days~~ in accordance with Surveillance Requirement 4.6.1.1.a.

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM

The OPERABILITY of the emergency feedwater (EFW) system ensures that the Reactor Coolant System can be cooled down to Shutdown Cooling (SDC) entry conditions from normal operating conditions in the event of a total loss of off-site power.

The EFW system is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the EFW system supplies sufficient water to cool the unit to SDC entry conditions, and steam is released through the ADVs.

A Note prohibits the application of LCO 3.0.4.b to an inoperable EFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an EFW train inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

SR 4.7.1.2.b.1 verifies that each EFW pump's developed head at the flow test point is greater than or equal to this required developed head. This test ensures that EFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by the ASME Operation and Maintenance (OM) Code. Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point that is indicative of pump overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing in accordance with the ASME OM Code satisfies this requirement. The SR for the turbine driven EFW pump is allowed to be deferred for up to 24 hours after exceeding 700 psia in the steam generators. This allowance will ensure the test is completed within a reasonable period of time after establishing sufficient steam pressure to perform the test.

SR 4.7.1.2.c ensures that EFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an EFAS signal. This is assured by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. The Surveillance~~18 month~~ f~~F~~requency controlled under the Surveillance Frequency Control Program is based on the need to perform the SRs under the conditions that apply during a unit outage and the potential for an unplanned transient if the SRs were performed with the reactor at power.

SR 4.7.1.2.d ensures that the EFW System is properly aligned by verifying the flow path from the condensate storage tank (CST) to each steam generator prior to entering MODE 2 operation, after more than 30 days below MODE 3. OPERABILITY of the EFW flow paths must be verified before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, and other administrative controls to ensure that flow paths remain OPERABLE. To further ensure EFW System alignment, the OPERABILITY of the flow paths is verified following extended outages to determine that no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION (CREVS) AND AIR CONDITIONING SYSTEM (CREACS) (continued)

SURVEILLANCE REQUIREMENTS

SR 4.7.6.1.1 a. and b.

These SRs, in conjunction with periodic preventative maintenance activities, provide verification that the CREACS will maintain the control room temperature within acceptable bound. In accordance with SR 4.7.6.1.1.a each train is verified to start and operate for at least 1 hour while maintaining Control Room temperature within the specified limit. The frequencies (controlled under the Surveillance Frequency Control Program~~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.

SR 4.7.6.1.2.a

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 starting the system from the control room and initiating flow through the HEPA filters and charcoal adsorbers. The CREVS is designed without heaters and need only be operated at least 15 minutes to demonstrate the function of the system. The ~~31-day~~Surveillance F~~f~~requency is controlled under the Surveillance Frequency Control Program and is based on the known reliability of the equipment and two train redundancy available.

SR 4.7.6.1.2.b

This SR verifies that upon injection of an actual or simulated control room high radiation test signal the Control Room automatically isolates within 10 seconds and the CREVS switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks. The Surveillance F~~f~~requency is controlled under the Surveillance Frequency Control Program~~of 18 months~~ and is based on industry operating experience and is consistent with the typical refueling cycle.

SR 4.7.6.1.2.c

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 4.7.6.1.2.d

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.

If one of the required DC electrical power subsystems is inoperable for reasons other than ACTION “a” (e.g., inoperable battery charger), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure could, however, result in the loss of the minimum necessary DC electrical subsystems to mitigate a worst case accident, continued power operation should not exceed 2 hours. The 2-hour AOT is based on Regulatory Guide (RG) 1.93 and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required AOT, 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 5 within the following 30 hours. The AOTs are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. The AOT to bring the unit to MODE 5 is consistent with the time required in RG 1.93.

Cascading to other TSs is not required solely due to a single station battery inoperability. In accordance with TS 3.0.5, the DC bus remains OPERABLE if its redundant power source (vital AC source via its battery charger) is OPERABLE and the redundant DC bus is fully OPERABLE (both vital AC and DC are available to supply the bus). Therefore, all DC loads associated with the affected train, including the respective EDG, remain OPERABLE. The 2-hour restoration period sufficiently takes into account the importance of the battery source and the vulnerability of supported equipment when a battery bank is out of service.

Surveillance Requirement (SR) 4.8.2.3.1 requires verifying battery terminal voltage while on float charge. This 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~~7-day~~ Frequency is consistent with manufacturer recommendations.

SR 4.8.2.3.2 verifies the design capacity of the chargers. According to Regulatory Guide 1.32, 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 SR Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during the~~se~~ Surveillance Frequency~~18 month intervals~~. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

SR 4.8.2.3.3 requires a battery service test, 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 ~~of 18 months~~ is consistent with the recommendations of RG 1.32 and RG 1.129, 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.

A performance discharge test may be performed in lieu of a service test. The performance discharge test required by SR 3.8.3.6 may be performed in lieu of the battery service test on a once per 60-month basis.

TS 3.8.2.4 DC Sources – Shutdown

In general, when the unit is shutdown, the TS requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1 and 2 have no specific analyses in MODES 3, 4, 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

With one battery with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of electrolyte level not met. Within 31 days the minimum established design limits for electrolyte level must be re-established, in accordance with ACTION “c”.

With electrolyte level below the top of the plates there is a potential for dryout and plate degradation. ACTION “c” addresses this potential (as well as provisions in Specification 6.5.15, Battery Monitoring and Maintenance Program). ACTIONs “c.i” and “c.ii” are modified by a Note that indicates they are only applicable if electrolyte level is below the top of the plates. Within 8 hours level is required to be restored to above the top of the plates and within 12 hours a visual inspection to very no leakage is performed. Specification 6.5.15.b.3 initiates action to equalize and test in accordance with manufacturer's recommendation following restoration of the electrolyte level to above the top of the plates. Based on the results of the manufacturer's recommended testing, the battery may have to be declared inoperable and the affected cell[s] replaced.

With one battery with pilot cell temperature less than the minimum established design limits, 12 hours is allowed to restore the temperature to within limits in accordance with ACTION “d”. A low electrolyte temperature limits the current and power available. Since the battery is sized with margin, while battery capacity is degraded, sufficient capacity exists to perform the intended function and the affected battery is not required to be considered inoperable solely as a result of the pilot cell temperature not met.

With both batteries with battery parameters not within limits there is not sufficient assurance that battery capacity has not been affected to the degree that the batteries can still perform their required function, given that redundant batteries are involved. This potential could result in a total loss of function on multiple systems that rely upon the batteries. The longer AOTs specified for battery parameters on non-redundant batteries not within limits are, therefore, not appropriate, and the parameters must be restored to within limits on at least one battery within 2 hours, in accordance with ACTION “e”.

With one or more batteries with any battery parameter outside the allowances of ACTION “a”, “b”, “c”, “d”, or “e”, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable. Additionally, discovering one battery with one or more battery cells float voltage less than 2.07 V and float current greater than 2 amps indicates that the battery capacity may not be sufficient to perform the intended functions. The battery must, therefore, be declared inoperable immediately.

Verifying battery float current while on float charge in accordance with SR 4.8.3.1 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~~7-day~~ Frequency, controlled under the Surveillance Frequency Control Program, is consistent with IEEE-450. The SR also 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 4.8.2.3.1. When this float voltage is not maintained, LCO 3.8.2.3 ACTION “a” is entered, 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.

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 6.5.15. SRs 4.8.3.2 and 4.8.3.5 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 Frequency for cell voltage verification ~~every 31 days~~ for pilot cell and ~~92 days~~ for each connected cell is consistent with IEEE-450.

The limit specified for electrolyte level in SR 4.8.3.3 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 Frequency is consistent with IEEE-450.

SR 4.8.3.4 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 Frequency is consistent with IEEE-450.

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. SR 4.8.3.6 is intended to determine overall battery degradation due to age and usage. The performance discharge test is acceptable for satisfying the battery service test requirements of SR 3.8.2.3.3 on a once per 60-month basis.

The acceptance criteria for this SR are consistent with IEEE-450, 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 SR Frequency for this test is in accordance with the Surveillance Frequency Control Program~~normally 60 months~~. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the SR Frequency is reduced to 12 months. However, if the battery shows no degradation, but has reached 85% of its expected life, the SR Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450, when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450.

ATTACHMENT 6 to

2CAN021802

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 Combustion Engineering Plants (NUREG-1432), to allow relocation of specific Arkansas Nuclear One, Unit 2 (ANO-2) 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 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 (SFCP). 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 (SFCP). 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 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 to

2CAN021802

ANO-2 TO NUREG 1432, REVISON 4, SR CROSS REFERENCE

Legend:

Arkansas Nuclear One, Unit 2 (ANO-2)Surveillance Requirement (SR) Frequency Identified for Relocation Not Included in TSTF-425 = Gray Row

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
Section 1.0, Definitions						
Definition 1.20	Definitions	Staggered Test Basis		yes	yes	N/A
Table 1.2	N/A	Frequency Notation	SFCP - In accordance with the Surveillance Frequency Control Program	no	yes	Millstone 2 St Lucie 1 and 2 Waterford 3
Section 3/4.1, Reactivity Control Systems						
4.1.1.1.1.b	N/A	When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.	at least once per 12 hours	no	yes	St Lucie 1 and 2 Waterford 3
4.1.1.1.1.e	3.1.1.1	When in MODES 3 or 4, at least once per 24 hours by consideration of at least the following factors: 1. Reactor coolant system boron concentration, 2. CEA position, 3. Reactor coolant system average temperature, 4. Fuel burnup based on gross thermal energy generation, 5. Xenon concentration, and 6. Samarium concentration.	at least once per 24 hours	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.1.1.1.2	3.1.2.1	Verify overall core reactivity balance is within + 1.0% Δk/k of predicted values in MODES 1, 2#, 3, and 4. The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within +1.0% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.	at least once per 31 Effective Full Power Days (EFPD)	yes	yes	N/A
4.1.1.2.b	3.1.1.1	At least once per 24 hours by consideration of at least the following factors: 1. Reactor coolant system boron concentration, 2. CEA position, 3. Reactor coolant system average temperature, 4. Fuel burnup based on gross thermal energy generation, 5. Xenon concentration, and 6. Samarium concentration.	At least once per 24 hours	yes	yes	N/A
4.1.1.3.a	N/A	Verifying at least one reactor coolant pump is in operation, or	within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration	no	yes	St Lucie 1 and 2
4.1.1.3.b	N/A	Verifying that at least one low pressure safety injection pump or containment spray pump is in operation as a shutdown cooling pump and supplying ≥ 2000 gpm through the reactor coolant system	within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration	no	yes	St Lucie 1 and 2

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.1.1.5	3.4.2.1	The Reactor Coolant System temperature (Tavg) shall be determined to be: ≥ 540 °F at least once per 12 hours.	at least once per 12 hours	yes	yes	N/A
4.1.3.1.1	3.1.4.1	The position of each CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group at least once per 12 hours.	at least once per 12 hours	yes	yes	N/A
4.1.3.1.2	3.1.4.3	Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 92 days.	at least once per 92 days	yes	yes	N/A
4.1.3.2	3.1.4.2	Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5 inches of each other at least once per 12 hours.	at least once per 12 hours	yes	yes	N/A
4.1.3.3	3.1.4.4	Each of the above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.	at least once per 18 months	yes	yes	N/A
4.1.3.4.c	N/A	At least once per 18 months	At least once per 18 months	no	yes	Millstone 2
4.1.3.5.b	3.1.5.1	At least once per 12 hours thereafter.	At least once per 12 hours thereafter	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.1.3.6	3.1.6.1 3.1.6.2	The position of each regulating CEA group and Group P CEAs shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Alarm is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups or Group P CEAs are inserted beyond the Long Term Steady State Insertion Limit or the Short Term Steady State Insertion Limit but within the Transient Insertion Limit shall be determined at least once per 24 hours	at least once per 12 hours at least once per 24 hours	yes	yes	N/A
N/A	3.1.6.3	{Verify PDIL alarm circuit is OPERABLE}	N/A	yes	no	N/A
N/A	3.1.7.1	{Verify part length CEA group position}	N/A	yes	no	N/A
Section 3/4.2, Power Distribution Limits						
4.2.1.2	3.2.1.1	The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.	at least once per 2 hours	yes	yes	N/A
4.2.1.3	3.2.1.2	At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on linear heat rate.	At least once per 31 days	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.2.2.2.b	3.2.2.1	At least once per 31 days of accumulated operation in MODE 1	At least once per 31 days of accumulated operation in MODE 1	yes	yes	N/A
4.2.3.b	3.2.3.1	Calculating the tilt at least once per 12 hours when the COLSS is inoperable	at least once per 12 hours	yes	yes	N/A
4.2.3.c	3.2.3.2	Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.	at least once per 31 days	yes	yes	N/A
4.2.3.d	3.2.3.3	Using the incore detectors at least once per 31 days to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.	at least once per 31 days	yes	yes	N/A
4.2.4.2	3.2.4.1	The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on any OPERABLE CPC channel, is within the limit specified in the CORE OPERATING LIMITS REPORT.	at least once per 2 hours	yes	yes	N/A
4.2.4.3	3.2.4.2	At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.	At least once per 31 days	yes	yes	N/A
4.2.5	3.4.1.3	The actual Reactor Coolant System total flow rate shall be determined to be within its limit at least once per 12 hours.	at least once per 12 hours	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.2.6	3.4.1.2	The Reactor Coolant Cold Leg Temperature shall be determined to be within its limit at least once per 12 hours.	at least once per 12 hours	yes	yes	N/A
4.2.7	3.2.5.1	The core average AXIAL SHAPE INDEX shall be determined to be within its limits at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.	at least once per 12 hours	yes	yes	N/A
4.2.8	3.4.1.1	The average pressurizer pressure shall be determined to be within its limit at least once per 12 hours.	at least once per 12 hours	yes	yes	N/A
N/A	3.4.1.4	{Verify by precision heat balance that RCS total flow rate is within the limits specified in the COLR}	N/A	yes	no	N/A
Section 3/4.3, Instrumentation						
4.3.1.1.1 Table 4.3-1	3.3.1.1 3.3.1.4 3.3.1.5 3.3.1.6 3.3.1.7 3.3.1.8 3.3.1.10 3.3.1.11 3.3.2.1 3.3.2.2 3.3.2.4 3.3.3.1 3.3.3.3 3.3.3.4 3.3.3.5 3.3.4.1 3.3.4.2 3.3.4.3 3.3.12.4	Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.	S/U - Prior to reactor startup S - At least once per 12 hours D - At last once per 24 hours M - At least once per 31 days Q - At least once per 92 days TA - At least once per 123 days R - At least once per 18 months	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
N/A	3.3.1.9	{Perform CHANNEL FUNCTIONAL TEST for Loss of Load Function}	N/A	yes	no	N/A
4.3.1.1.2	3.3.1.13 3.3.2.3 3.3.2.4	The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.	prior to each reactor startup unless performed during the preceding 92 days at least once per 18 months	no	yes (18-month frequency only)	Millstone 2 St Lucie 1 and 2 Waterford 3
4.3.1.1.3	3.3.1.14 3.3.2.5	The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.	at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the “Total No. of Channels” column of Table 3.3-1	yes	yes	N/A
4.3.1.1.4	3.3.1.3	The Core Protection Calculator System shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours	at least once per 12 hours	yes	yes	N/A
Table 4.3-1 Note 2	3.3.1.4	a. Between 15% and 80% of RATED THERMAL POWER, compare the Linear Power Level, the CPC ΔT power, and the CPC nuclear power signals to the calorimetric calculation. b. At or above 80% of RATED THERMAL POWER, compare the Linear Power Level, the CPC ΔT power, and CPC nuclear power signals to the calorimetric calculation.	D - At least once per 24 hours	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
Table 4.3-1 Note 3	3.3.1.6	Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.	M - At least once per 31 days	yes	yes	N/A
Table 4.3-1 Note 7	3.3.1.2 3.3.1.5	Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and, if necessary, adjust the CPC flow calibration addressable constant FC1 such that each CPC indicated flow is less than or equal to the measured flow rate.	S - At least once per 12 hours	yes	yes	N/A
Table 4.3-1 Note 9	3.3.1.7	The CPC CHANNEL FUNCTIONAL TEST shall include the verification that the correct values of addressable constants are installed in each OPERABLE CPC.	TA - At least once per 123 days	yes	yes	N/A
Table 4.3-1 Note 10	N/A	Staggered Test Bases (Applicable to Table 4.3-1 Channel Functional Tests with a frequency of TA)	TA - At least once per 123 days (Note 10) on a Staggered Test Bases	no	yes	Waterford 3 (Table 4.3-2 Notes)
N/A	3.3.3.2	{Check the CEAC auto restart count}	N/A	yes	no	N/A
N/A	3.3.3.6	{Verify the isolation characteristics of each CEAC isolation amplifier and each optical isolator for CEAC to CPC data transfer}	N/A	yes	no	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.3.2.1.1 Table 4.3-2	3.3.5.1 3.3.5.2 3.3.5.3 3.3.6.1 3.3.6.2 3.3.6.3 3.3.7.1 3.3.7.2 3.3.7.3	Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.	S - At least once per 12 hours TA - At least once per 123 days R - At least once per 18 months	yes	yes	N/A
4.3.2.1.2	3.3.5.3 3.3.5.5	The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.	at least once per 18 months	yes	yes	N/A
4.3.2.1.3	3.3.5.4	The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.	at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
Table 4.3-2 Note 2	N/A	Staggered Test Bases (Applicable to Table 4.3-2 Channel Functional Tests with a frequency of TA)	TA - At least once per 123 days (Note 2) on a Staggered Test Bases	no	yes	Waterford 3
4.3.3.1 Table 4.3-3	3.3.8.1 3.3.8.2 3.3.8.3 3.3.8.4 3.3.8.5 3.3.8.6 3.3.9.1 3.3.9.2 3.3.9.3 3.3.9.4 3.9.3.2	Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.	S - At least once per 12 hours M - At least once per 31 days R - At least once per 18 months Also - prior to cntmt purge operation	yes	yes	N/A
N/A	3.3.8.7	{Verify that response time of required CPIS channel is within limits}	N/A	yes	no	N/A
N/A	3.3.8.8	{Perform CHANNEL FUNCTIONAL TEST on required CPIS Manual Trip channel}	N/A	yes	no	N/A
N/A	3.3.10.1	{Perform a CHANNEL CHECK on required FHIS radiation monitor channel}	N/A	yes	no	N/A
N/A	3.3.10.2	{Perform a CHANNEL FUNCTIONAL TEST on required FHIS radiation monitor channel. Verify radiation monitor setpoint [Allowable Values]:}	N/A	yes	no	N/A
N/A	3.3.10.3	{Perform a CHANNEL FUNCTIONAL TEST on required FHIS Actuation Logic channel}	N/A	yes	no	N/A
N/A	3.3.10.4	{Perform a CHANNEL FUNCTIONAL TEST on required FHIS Manual Trip logic}	N/A	yes	no	N/A
N/A	3.3.10.5	{Perform a CHANNEL CALIBRATION on required FHIS radiation monitor channel}	N/A	yes	no	N/A
N/A	3.3.10.6	{[Verify response time of required FHIS channel is within limits]}	N/A	yes	no	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.3.3.5 Table 4.3-6	3.3.12.1 3.3.12.3	Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.	M - At least once per 31 days R - At least once per 18 months	yes	yes	N/A
N/A	3.3.12.2	{Verify each required control circuit and transfer switch is capable of performing the intended function}	N/A	yes	no	N/A
4.3.3.6 Table 4.3-10	3.3.11.1 3.3.11.2	Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-10.	M - At least once per 31 days R - At least once per 18 months	yes	yes	N/A
N/A	3.3.13.1	{Perform CHANNEL CHECK of the [Logarithmic] Power Monitoring Channels}	N/A	yes	no	N/A
N/A	3.3.13.2	{Perform CHANNEL FUNCTIONAL TEST}	N/A	yes	no	N/A
N/A	3.3.13.3	{Perform CHANNEL CALIBRATION}	N/A	yes	no	N/A
Section 3/4.4, Reactor Coolant System						
4.4.1.1	3.4.4.1	The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.	at least once per 12 hours.	yes	yes	N/A
4.4.1.2.1	3.4.5.3	At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.	once per 7 days	yes	yes	N/A
4.4.1.2.2	3.4.5.1	At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.	at least once per 12 hours.	yes	yes	N/A
N/A	3.4.5.2	{Verify secondary side water level in each steam generator ≥ [25]%}	N/A	yes	no	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.4.1.3.2	3.4.6.3 3.4.7.3 3.4.8.2	The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability	once per 7 days	yes	yes	N/A
4.4.1.3.3	3.4.6.2 3.4.7.2	The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be ≥ 23% indicated level at least once per 12 hours.	at least once per 12 hours	yes	yes	N/A
4.4.1.3.4	3.4.6.1 3.4.7.1 3.4.8.1	At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.	at least once per 12 hours.	yes	yes	N/A
4.4.4.1	3.4.9.1	The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.	at least once per 12 hours	yes	yes	N/A
4.4.4.2(a)	3.4.9.3	At least once per 12 hours by verifying emergency power is available to the heater groups, and	at least once per 12 hours	yes	yes	N/A
4.4.4.2(b)	3.4.9.2	At least once per 18 months by verifying that the summed power consumption of the two proportional heater groups is ≥ 150 KW.	At least once per 18 months	yes	yes	N/A
4.4.6.1.a	3.4.15.1	Performing a CHANNEL CHECK of the required containment atmosphere radioactivity monitors at least once per 12 hours.	at least once per 12 hours	yes	yes	N/A
4.4.6.1.b	N/A	Performing a CHANNEL CHECK of the containment sump level monitor at least once per 12 hours.	at least once per 12 hours	no	yes	Waterford 3
4.4.6.1.c	3.4.15.2	Performing a CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitors at least once per 31 days.	at least once per 31 days.	yes	yes	N/A
4.4.6.1.d	3.4.15.3	Performing a CHANNEL CALIBRATION of the containment sump level monitor at least once per 18 months.	at least once per 18 months	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.4.6.1.e	3.4.15.4	Performing a CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitors at least once per 18 months.	at least once per 18 months	yes	yes	N/A
N/A	3.4.15.5	{[Perform CHANNEL CALIBRATION of the required containment air cooler condensate flow rate monitor]}	N/A	yes	no	N/A
4.4.6.2.1.a	3.4.13.1	Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode	at least once per 72 hours	yes	yes	N/A
4.4.6.2.1.b	N/A	Monitoring the reactor head flange leakoff temperature at least once per 24 hours.	at least once per 24 hours	no	yes	St Lucie 1 and 2
4.4.6.2.2	3.4.13.2	Primary to secondary leakage shall be verified to be ≤ 150 gallons per day through any one SG at least once per 72 hours*.	at least once per 72 hours	yes	yes	N/A
N/A	3.4.14.2	{Verify SDC System autoclosure interlock prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ [425] psig}	N/A	yes	no	N/A
N/A	3.4.14.3	{Verify SDC System autoclosure interlock causes the valves to close automatically with a simulated or actual RCS pressure signal ≥ [600] psig}	N/A	yes	no	N/A
4.4.8.1	3.4.16.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 3100 µCi/gm once every 7 days.	once every 7 days	yes	yes	N/A
4.4.8.2.a	3.4.16.2	once every 14 days, and	once every 14 days	yes	yes	N/A
N/A	3.4.16.3	{Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours}	N/A	yes	no	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.4.9.1.1	3.4.3.1	The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.	at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations	yes	yes	N/A
4.4.11	N/A	Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by verifying flow through the reactor coolant vent system vent paths.	at least once per 18 months	no	yes	St Lucie 1 and 2 Waterford 3
N/A	3.4.11.1	{Perform a complete cycle of each block valve}	N/A	yes	no	N/A
N/A	3.4.11.2	{Perform a complete cycle of each PORV}	N/A	yes	no	N/A
N/A	3.4.11.3	{[Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems]}	N/A	yes	no	N/A
N/A	3.4.11.4	{[Verify PORVs and block valve(s) are capable of being powered from an emergency power supply]}	N/A	yes	no	N/A
4.4.12.1	3.4.12.5	Verify both sets of LTOP relief valve isolation valves are open at least once per 72 hours when the LTOP relief valves are being used for overpressure protection.	at least once per 72 hours when the LTOP relief valves are being used for overpressure protection	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.4.12.2	3.4.12.4	The RCS vent path shall be verified to be open at least once per 12 hours** when the vent path is being used for overpressure protection. **Except when the vent path is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify this valve is open at least once per 31 days.	at least once per 12 hours when the vent path is being used for overpressure protection. at least once per 31 days when the vent path is provided with a valve which is locked, sealed, or otherwise secured in the open position.	yes	yes	N/A
4.4.12.3	3.4.12.3	Verify that each SIT [safety injection tank] is isolated, when required, once every 12 hours.	at least once per 12 hours	yes	yes	N/A
N/A	3.4.12.1	{Verify a maximum of one HPSI pump is capable of injecting into the RCS}	N/A	yes	no	N/A
N/A	3.4.12.2	{Verify a maximum of one charging pump is capable of injecting into the RCS}	N/A	yes	no	N/A
N/A	3.4.12.6	{Perform CHANNEL FUNCTIONAL TEST on each required PORV, excluding actuation}	N/A	yes	no	N/A
N/A	3.4.12.7	{Perform CHANNEL CALIBRATION on each required PORV actuation channel}	N/A	yes	no	N/A
Section 3/4.5, Emergency Core Cooling Systems (ECCS)						
4.5.1.a.1	3.5.1.2 3.5.1.3	At least once per 12 hours by: Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and	at least once per 12 hours	yes	yes	N/A
4.5.1.a.2	3.5.1.1	At least once per 12 hours by: Verifying that each safety injection tank isolation valve (2CV-5003, 2CV-5023, 2CV-5043 and 2CV-5063) is open.	at least once per 12 hours	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.5.1.b	3.5.1.4	At least once per 31 days and within 6 hours after each solution volume increase of ≥ 5% of indicated tank level that is not the result of addition from the RWT, by verifying the boron concentration of the safety injection tank solution.	at least once per 31 day and within 6 hours after each solution volume increase of ≥ 5% of indicated tank level that is not the result of addition from the RWT	yes	yes (31 day Frequency only)	N/A
4.5.1.c	3.5.1.5	At least once per 31 days when the RCS pressure is above 2000 psia, by verifying that power to the isolation valve operator is removed by maintaining the motor circuit breaker open under administrative control.	at least once per 31 days when the RCS pressure is above 2000 psia	yes	yes	N/A
4.5.1.d	N/A	At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions: 1. When the RCS pressure exceeds 700 psia, and 2. Upon receipt of a safety injection test signal.	at least once per 18 months	no	yes	St Lucie 1 and 2 Waterford 3
4.5.2.a	3.5.2.1	At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:	at least once per 12 hours	yes	yes	N/A
4.5.2.b	3.5.2.2	At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position	at least once per 31 days	yes	yes	N/A
N/A	3.5.2.3	{[Verify ECCS piping is full of water.]}	N/A	yes	no	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.5.2.d	3.5.2.10	At least once per 18 months by a visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.	at least once per 18 months	yes	yes	N/A
4.5.2.e.1	3.5.2.6	Verifying that each automatic valve in the flow path actuates to its correct position on SIAS and RAS test signals.	at least once per 18 months	yes	yes	N/A
4.5.2.e.2	3.5.2.7	Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal: a. High-Pressure Safety Injection pump. b. Low-Pressure Safety Injection pump.	at least once per 18 months	yes	yes	N/A
4.5.2.g	3.5.2.9	At least once per 18 months by verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves: LPSI System Valve Number a. 2CV-5037-1 b. 2CV-5017-1 c. 2CV-5077-2 d. 2CV-5057-2	at least once per 18 months	yes	yes	N/A
N/A	3.5.2.8	{Verify each LPSI pump stops on an actual or simulated actuation signal}	N/A	yes	no	N/A
4.5.4.a.1	3.5.4.2	Verifying the contained borated water volume in the tank	at least once per 7 days	yes	yes	N/A
4.5.4.a.2	3.5.4.3	Verifying the boron concentration of the water.	at least once per 7 days	yes	yes	N/A
4.5.4.b	3.5.4.1	At least once per 24 hours by verifying the RWT temperature.	at least once per 24 hours	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
Section 3/4.6, Containment Systems						
4.6.1.1.a	3.6.3.3	At least once per 31 days by verifying that each containment isolation manual valve and blind flange (Note 1) that is located outside containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative control as permitted by Specification 3.6.3.1	at least once per 31 days	yes	yes	N/A
4.6.1.3.2	3.6.2.2	Each containment air lock interlock shall be demonstrated OPERABLE by testing the air lock interlock mechanism at least once per 184 days.	at least once per 184 days	yes	yes	N/A
4.6.1.4	3.6.4.1 3.6.5.1	The primary containment internal pressure and average air temperature shall be determined to be within the limits at least once per 12 hours. The containment average air temperature shall be the temperature of the air in the containment HVAC common return air duct upstream of the fan/cooler units.	at least once per 12 hours	yes	yes	N/A
4.6.1.6	3.6.3.1 3.6.3.2	The containment purge supply and exhaust isolation valves shall be determined closed at least once per 31 days.	at least once per 31 days	yes	yes	N/A
4.6.2.1.a.1	3.6.6A.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	at least once per 31 days	yes	yes	N/A
4.6.2.1.a.2	3.6.6A.4	Verifying that the system piping is full of water from the RWT to at least elevation 505' (equivalent to > 12.5% indicated narrow range level) in the risers within the containment.	at least once per 31 days	yes	yes	N/A
4.6.2.1.c.1	3.6.6A.6	Verifying that each automatic valve in the flow path actuates to its correct position on CSAS and RAS test signals.	at least once per 18 months	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.6.2.1.c.2	N/A	Verifying that upon a RAS test signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.	at least once per 18 months	no	yes	St Lucie 1 and 2 Waterford 3
4.6.2.1.c.3	3.6.6A.7	Verifying that each spray pump starts automatically on a CSAS test signal.	at least once per 18 months	yes	yes	N/A
4.6.2.2.a	3.5.5.1	At least once per 18 months by verifying that the buffering agent baskets contain ≥ 308 ft^3 of NaTB decahydrate.	at least once per 18 months	yes	yes	St Lucie 1 and 2 Waterford 3
4.6.2.2.b	3.5.5.2	At least once per 18 months by verifying that a sample from the buffering agent baskets provides adequate pH adjustment of borated water.	at least once per 18 months	yes	yes	St Lucie 1 and 2 Waterford 3
4.6.2.3.a.1	3.6.6A.3	Verifying that service water flow rate to the group of cooling units is ≥ 1250 gpm and that each group has two operable fans.	At least once per 14 days	yes	yes	N/A
4.6.2.3.a.2	N/A	Addition of a biocide to the service water during the surveillance in 4.6.2.3.a.1 above, whenever service water temperature is between 60 °F and 80 °F.	At least once per 14 days	no	yes	N/A
4.6.2.3.b.1	N/A	Starting (unless already operating) each operational cooling unit from the control room.	at least once per 31 days	no	yes	Millstone 2 St Lucie 1 and 2 Waterford 3
4.6.2.3.b.2	3.6.6A.2	Verifying that each operational cooling unit operates for at least 15 minutes.	at least once per 31 days	yes	yes	N/A
4.6.2.3.c	3.6.6A.8	At least once per 18 months by verifying that each cooling unit starts automatically on a CCAS test signal.	at least once per 18 months	yes	yes	N/A
4.6.3.1.2	3.6.3.7	Each containment isolation valve shall be demonstrated OPERABLE at least once per 18 months by verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position.	at least once per 18 months	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
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	N/A
N/A	3.6.7.1	{Verify each spray additive 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}	N/A	yes	no	N/A
N/A	3.6.7.2	{Verify spray additive tank solution volume is ≥ [816] gal [90%] and ≤ [896] gal [100%]}	N/A	yes	no	N/A
N/A	3.6.7.3	{Verify spray additive tank [N2H4] solution concentration is ≥ [33]% and ≤ [35]% by weight}	N/A	yes	no	N/A
N/A	3.6.7.5	{Verify each spray additive 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
N/A	3.6.7.6	{[Verify spray additive flow [rate] from each solution's flow path.]}	N/A	yes	no	N/A
N/A	3.6.9.1	{Operate each HMS train for ≥ 15 minutes}	N/A	yes	no	N/A
N/A	3.6.9.2	{Verify each HMS train flow rate on slow speed is ≥ [37,000] cfm}	N/A	yes	no	N/A
N/A	3.6.9.3	{Verify each HMS train starts on an actual or simulated actuation signal}	N/A	yes	no	N/A
N/A	3.6.10.1	{Operate each ICS train for [≥ 10 continuous hours with heaters operating or (for systems without heaters) ≥ 15 minutes]}	N/A	yes	no	N/A
N/A	3.6.10.3	{Verify each ICS train actuates on an actual or simulated actuation signal}	N/A	yes	no	N/A
N/A	3.6.10.4	{[Verify each ICS filter bypass damper can be opened.]}	N/A	yes	no	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
Section 3/4.7, Plant Systems						
4.7.1.2.a.1	3.7.5.1	At least once per 31 days by: Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.	at least once per 31 days	yes	yes	N/A
4.7.1.2.c.1	3.7.5.3	Verifying that each automatic valve in the flow path actuates to its correct position on MSIS or EFAS test signals.	at least once per 18 months	yes	yes	N/A
4.7.1.2.c.2	3.7.5.4	Verifying that the motor driven pump starts automatically upon receipt of an EFAS test signal.	at least once per 18 months	yes	yes	N/A
4.7.1.2.c.3	3.7.5.3	Verifying that the turbine driven pump steam supply MOV opens automatically upon receipt of an EFAS test signal.	at least once per 18 months	yes	yes	N/A
4.7.1.3.1	3.7.6.1	The above required condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the emergency feedwater pumps.	at least once per 12 hours	yes	yes	N/A
4.7.1.3.2	3.7.8.1 3.7.8.2	The service water system shall be demonstrated OPERABLE at least once per 12 hours by verifying that at least one service water loop is operating and that the service water system - emergency feedwater system isolation valves are either open or OPERABLE whenever the service water system is the supply source for the emergency feedwater pumps.	at least once per 12 hours	yes	yes	N/A
4.7.1.4.1 Table 4.7-2	3.7.19.1	Gross Activity Determination - At least once per 72 hours	at least once per 72 hours	yes	yes	N/A
4.7.1.4.2.a Table 4.7-2	3.7.19.1	1 per 31 days, whenever the gross activity determination is greater than 10% of the allowable iodine limit.	once per 31 days	yes	yes	Waterford 3

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.7.1.4.2.b Table 4.7-2	3.7.19.1	1 per 6 months, whenever the gross activity determination is below 10% of the allowable iodine limit.	once per 6 months	yes	yes	Millstone 2 Waterford 3
N/A	3.7.2.2	{Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal}	N/A	yes	no	N/A
N/A	3.7.3.2	{Verify each MFIV [and [MFIV] bypass valve] actuates to the isolation position on an actual or simulated actuation signal}	N/A	yes	no	N/A
N/A	3.7.4.1	{Verify one complete cycle of each ADV}	N/A	yes	no	N/A
N/A	3.7.4.2	{[Verify one complete cycle of each ADV block valve.]}	N/A	yes	no	N/A
4.7.2.1	N/A	The pressure in each side of the steam generators shall be determined to be < 275 psig at least once per hour when the temperature of either the primary or secondary coolant is < 90°F.	at least once per hour when the temperature of either the primary or secondary coolant is < 90°F	no	yes	N/A
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	N/A
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
N/A	3.7.7.3	{Verify each CCW pump starts automatically on an actual or simulated actuation signal}	N/A	yes	no	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.7.3.1.a	3.7.8.1	At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.	at least once per 31 days	yes	yes	N/A
4.7.3.1.b	3.7.8.2	At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on CCAS, MSIS and RAS test signals.	at least once per 18 months	yes	yes	N/A
N/A	3.7.8.3	{Verify each SWS pump starts automatically on an actual or simulated actuation signal}	N/A	yes	no	N/A
4.7.4.1.a	3.7.9.1	At least once per 24 hours by verifying that the indicated water level of the ECP is greater than or equal to that required for an ECP volume of 70 acre-feet.	at least once per 24 hours	yes	yes	N/A
4.7.4.1.b	3.7.9.2	At least once per 24 hours during the period of June 1 through September 30 by verifying that the pond's average water temperature at the point of discharge from the pond is within its limit.	at least once per 24 hours	yes	yes	N/A
N/A	3.7.9.3	{[Operate each cooling tower fan for ≥ [15] minutes]}	N/A	yes	no	N/A
4.7.4.1.c.1	3.7.9.1	A contained water volume of ECP ≥ 70 acre-feet, and	At least once per 12 months	yes	yes	N/A
4.7.4.1.c.2	3.7.9.1	The minimum indicated water level needed to ensure a volume of 70 acre-feet is maintained.	At least once per 12 months	yes	yes	N/A
4.7.4.1.d	N/A	At least once per 12 months by performance of a visual inspection of the ECP to verify conformance with design requirements.	At least once per 12 months	no	yes	N/A
4.7.5.1.a	N/A	Measurement at least once per 24 hours when the water level is below elevation 350 feet Mean Sea Level USGS datum, and	at least once per 24 hours when the water level is below elevation 350 feet	no	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.7.5.1.b	N/A	Measurement at least once per 2 hours when the water level is equal to or above elevation 350 feet Mean Sea Level USGS datum.	at least once per 2 hours when the water level is equal to or above elevation 350 feet Mean Sea Level USGS datum	no	yes	N/A
N/A	3.7.10.1	{Verify each ECW [Essential Chilled Water] 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}	N/A	yes	no	N/A
N/A	3.7.10.2	{Verify the proper actuation of each ECW System component on an actual or simulated actuation signal}	N/A	yes	no	N/A
4.7.6.1.1.a.1	N/A	Starting each unit from the control room, and	at least once per 31 days	no	yes	Millstone 2 St Lucie 1 and 2 Waterford 3
4.7.6.1.1.a.2	3.7.12.1	Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤ 84°F D.B.	at least once per 31 days	yes	yes	N/A
4.7.6.1.1.b	N/A	At least once per 18 months by verifying a system flow rate of 9900 cfm ± 10%.	at least once per 18 months	no	yes	Millstone 2 Waterford 3
4.7.6.1.2.a	3.7.11.1	At least once per 31 days by verifying that the system operates for at least 15 minutes	at least once per 31 days	yes	yes	N/A
4.7.6.1.2.b	3.3.9.5 3.3.9.6 3.7.11.3	At least once per 18 months by verifying that on a control room high radiation signal, either actual or simulated, the system automatically isolates the control room and switches into a recirculation mode of operation.	at least once per 18 months	yes	yes	N/A
N/A	3.7.13.1	{Operate each ECCS PREACS train for [≥ 10 continuous hours with the heater operating or (for systems without heaters) ≥ 15 minutes]}	N/A	yes	no	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
N/A	3.7.13.3	{Verify each ECCS PREACS train actuates on an actual or simulated actuation signal}	N/A	yes	no	N/A
N/A	3.7.13.4	{Verify one ECCS PREACS train can maintain a negative pressure ≥ [ ] inches water gauge relative to atmospheric pressure during the [post accident] mode of operation at a flow rate of ≤ [20,000] cfm}	N/A	yes	no	N/A
N/A	3.7.13.5	{[Verify each ECCS PREACS filter bypass damper can be opened}	N/A	yes	no	N/A
N/A	3.7.14.1	{Operate each FBACS train for [≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes]}	N/A	yes	no	N/A
N/A	3.7.14.3	{[Verify each FBACS train actuates on an actual or simulated actuation signal.]}	N/A	yes	no	N/A
N/A	3.7.14.4	{Verify one FBACS train can maintain a negative pressure ≥ [ ] inches water gauge with respect to atmospheric pressure, during the [post accident] mode of operation at a flow rate ≤ [3000] cfm}	N/A	yes	no	N/A
N/A	3.7.14.5	{[Verify each FBACS filter bypass damper can be opened.]}	N/A	yes	no	N/A
N/A	3.7.15.1	{Operate each PREACS train for [≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes]}	N/A	yes	no	N/A
N/A	3.7.15.3	{[Verify each PREACS train actuates on an actual or simulated actuation signal.]}	N/A	yes	no	N/A
N/A	3.7.15.4	{Verify one PREACS train can maintain a negative pressure ≥ [ ] inches water gauge with respect to atmospheric pressure, during the [post accident] mode of operation at a flow rate ≤ [3000] cfm}	N/A	yes	no	N/A
N/A	3.7.15.5	{[Verify each PREACS filter bypass damper can be opened.]}	N/A	yes	no	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.7.9.1.2.a.1	N/A	a. Sources in use - At least once per six months for all sealed sources containing radioactive material: 1. With a half-life greater than 30 days (excluding Hydrogen 3)	At least once per six months	no	yes	St Lucie 1 and 2
4.7.9.1.2.a.2	N/A	a. Sources in use - At least once per six months for all sealed sources containing radioactive material: 2. In any form other than gas	At least once per six months	no	yes	St Lucie 1 and 2
Section 3/4.8, Electrical Power Systems						
4.8.1.1.1.a	3.8.1.1	Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and	at least once per 7 days	yes	yes	N/A
4.8.1.1.1.b	3.8.1.8	Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.	at least once per 18 months	yes	yes	N/A
4.8.1.1.2.a.1	3.8.1.4	Verifying the fuel level in the day fuel tank.	At least once per 31 days on a STAGGERED TEST BASIS	yes	yes	N/A
N/A	3.8.1.5	{Check for and remove accumulated water from each day tank [and engine mounted tank]}	N/A	yes	no	N/A
4.8.1.1.2.a.3	3.8.1.6	Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.	At least once per 31 days on a STAGGERED TEST BASIS	yes	yes	N/A
4.8.1.1.2.a.4	3.8.1.2 3.8.1.7	Verifying the diesel starts from a standby condition and accelerates to at least 900 rpm in < 15 seconds.	At least once per 31 days on a STAGGERED TEST BASIS	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.8.1.1.2.a.5	3.8.1.3	Verifying the generator is synchronized, loaded to an indicated 2600 to 2850 Kw and operates for ≥ 60 minutes.	At least once per 31 days on a STAGGERED TEST BASIS	yes	yes	N/A
4.8.1.1.2.a.6	N/A	Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.	At least once per 31 days on a STAGGERED TEST BASIS	no	yes	St Lucie 1 and 2 Waterford 3
4.8.1.1.2.c.2	3.8.1.18	Verifying during shutdown that the automatic sequence time delay relays are OPERABLE at their setpoint ± 10% of the elapsed time for each load block.	at least once per 18 months	yes	yes	N/A
4.8.1.1.2.c.3	3.8.1.9	Verifying during shutdown the generator capability to reject a load of greater than or equal to its associated single largest post-accident load, and maintain voltage at 4160 ± 500 volts and frequency at 60 ± 3 Hz.	at least once per 18 months	yes	yes	N/A
4.8.1.1.2.c.4	3.8.1.10	Verifying during shutdown the generator capability to reject a load of 2850 Kw without exceeding 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal, whichever is lower.	at least once per 18 months	yes	yes	N/A
4.8.1.1.2.c.5	3.8.1.11	Simulating during shutdown a loss of offsite power by itself, and: a. Verifying de-energization of the emergency busses and load shedding from the emergency busses. b. Verifying the diesel starts from a standby condition on the undervoltage auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected shutdown loads through the time delay relays and operates for ≥ 5 minutes while its generator is loaded with the shutdown loads.	at least once per 18 months	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.8.1.1.2.c.6	3.8.1.12	Verifying during shutdown that on a Safety Injection Actuation Signal (SIAS) actuation test signal (without loss of offsite power) the diesel generator starts on the auto-start signal and operates on standby for ≥ 5 minutes.	at least once per 18 months	yes	yes	N/A
4.8.1.1.2.c.7	3.8.1.13	Verifying during shutdown that all diesel generator trips, except engine overspeed, lube oil pressure, generator differential, and engine failure to start, are automatically bypassed upon a Safety Injection Actuation Signal.	at least once per 18 months	yes	yes	N/A
4.8.1.1.2.c.8	3.8.1.19	Simulating during shutdown a loss of offsite power in conjunction with SIAS and: a) Verifying de-energization of the emergency busses and load shedding from the emergency busses. b) Verifying the diesel starts from a standby condition on the auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency (accident) loads through the Time Delay Relays and operates for ≥ 5 minutes while its generator is loaded with the emergency loads.	at least once per 18 months	yes	yes	N/A
4.8.1.1.2.c.9	3.8.1.14 3.8.1.15	Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to an indicated 3000 to 3200 Kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 2600 to 2850 Kw. Within 5 minutes after completing this 24 hour test, perform 4.8.1.1.2.a.4.	at least once per 18 months	yes	yes	N/A
4.8.1.1.2.c.10	3.8.1.19	Verifying that the auto-connected loads to each diesel generator do not exceed the 2 hour rating of 3135 Kw.	at least once per 18 months	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.8.1.1.2.c.11	3.8.1.16	Verifying during shutdown the diesel generator's capability to: a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power, b) Transfer its loads to the offsite power source, and c) Proceed through its shutdown sequence.	at least once per 18 months	yes	yes	N/A
4.8.1.1.2.c.12	3.8.1.17	Verifying during shutdown that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the auto-connected emergency (accident) loads with offsite power.	at least once per 18 months	yes	yes	N/A
4.8.1.1.2.c.13	3.8.1.6	Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.	at least once per 18 months	yes	yes	N/A
4.8.1.1.2.d	3.8.1.20	At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 900 rpm in ≤ 15 seconds.	At least once per 10 years or after any modifications which could affect diesel generator interdependence	yes	yes	N/A
4.8.1.3.1	3.8.3.1	At least once per 31 days on a STAGGERED TEST BASIS verify the fuel oil storage tank contains ≥ 22,500 gallons of fuel.	At least once per 31 days on a STAGGERED TEST BASIS	yes	yes	N/A
N/A	3.8.3.2	{Verify lubricating oil inventory is ≥ a [7] day supply}	N/A	yes	no	N/A
N/A	3.8.3.4	{Verify each DG air start receiver pressure is ≥ [225] psig}	N/A	yes	no	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
N/A	3.8.3.5	{Check for and remove accumulated water from each fuel oil storage tank}	N/A	yes	no	N/A
4.8.2.1	3.8.9.1	The specified A.C. busses shall be determined OPERABLE with tie breakers open between redundant busses at least once per 7 days by verifying correct breaker alignment and indicated power availability.	at least once per 7 days	yes	yes	N/A
4.8.2.2	3.8.10.1	The specified A.C. busses shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.	at least once per 7 days	yes	yes	N/A
4.8.2.3.1	3.8.4.1	At least once per 7 days by verifying that the battery terminal voltage is greater than or equal to the minimum established float voltage.	at least once per 7 days	yes	yes	N/A
4.8.2.3.2	3.8.4.2	At least once per 18 months by verifying that each battery charger supplies ≥ 300 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours or, by verifying that 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.	at least once per 18 months	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.8.2.3.3	3.8.4.3	At least once per 18 months by verifying that the battery capacity is adequate to supply, and maintain in OPERABLE status, required emergency loads for the design duty cycle when subjected to a battery service test. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this Surveillance. The battery performance discharge test required by Surveillance Requirement 4.8.3.6 may be performed in lieu of the battery service test once per 60 months.	at least once per 18 months	yes	yes	N/A
4.8.2.4.1	3.8.10.1	The above required 125-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.	at least once per 7 days	yes	yes	N/A
4.8.3.1	3.8.6.1	At least once per 7 days by verifying that each battery float current is ≤ 2 amps. This Surveillance is not required when battery terminal voltage is less than the minimum established float voltage of Surveillance Requirement 4.8.2.3.1.	at least once per 7 days	yes	yes	N/A
4.8.3.2	3.8.6.2	At least once per 31 days by verifying that each battery pilot cell float voltage is ≥ 2.07 V.	at least once per 31 days	yes	yes	N/A
4.8.3.3	3.8.6.3	At least once per 31 days by verifying that each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	at least once per 31 days	yes	yes	N/A
4.8.3.4	3.8.6.4	At least once per 31 days by verifying that each battery pilot cell temperature is greater than or equal to minimum established design limits.	at least once per 31 days	yes	yes	N/A
4.8.3.5	3.8.6.5	At least once per 92 days by verifying that each battery connected cell float voltage is ≥ 2.07 V.	At least once per 92 days	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.8.3.6	3.8.6.6	At least once per 60 months by verifying the battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this Surveillance. In addition to the 60-month test interval, the performance discharge test shall be performed:	At least once per 60 months	yes	yes	N/A
N/A	3.8.7.1	{Verify correct inverter voltage, [frequency,] and alignment to required AC vital buses}	N/A	yes	no	N/A
N/A	3.8.8.1	{Verify correct inverter voltage, [frequency,] and alignment to required AC vital buses}	N/A	yes	no	N/A
Section 3/4.9, Refueling Operations						
4.9.1.2	3.9.1.1	The boron concentration of the reactor coolant and the refueling canal shall be determined by chemical analysis at least once per 72 hours.	at least once per 72 hours	yes	yes	N/A
4.9.2.a	3.9.2.1	A CHANNEL CHECK at least once per 12 hours,	at least once per 12 hours	yes	yes	N/A
4.9.2.b	N/A	A CHANNEL FUNCTIONAL TEST at least once per 7 days, and	at least once per 7 days	no	yes	St Lucie 1 and 2 Waterford 3
N/A	3.9.2.2	{Perform CHANNEL CALIBRATION [(SRMs)]}	N/A	yes	no	N/A
4.9.4.1	3.9.3.1	Each of the above required containment penetrations shall be determined to be in its above required conditions within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment.	within 72 hours prior to the start at least once per 7 days	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.9.5	N/A	Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.	within one hour prior to the start at least once per 12 hours	no	yes	St Lucie 1 and 2
4.9.7	N/A	The crane electrical power disconnect which prevents crane travel over the spent fuel pool shall be verified open under administrative control at least once per 7 days, or the crane travel interlock which prevents crane travel over the spent fuel pool shall be demonstrated OPERABLE within 4 hours prior to each use of the crane for lifting loads in excess of 2000 pounds	at least once per 7 days	no	yes	N/A
4.9.8.1	3.9.4.1	A shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of ≥ 2000 gpm at least once per 24 hours.	at least once per 24 hours	yes	yes	N/A
N/A	3.9.5.1	{Verify required SDC loops are OPERABLE and one SDC loop is in operation}	N/A	yes	no	N/A
N/A	3.9.5.2	{Verify correct breaker alignment and indicated power available to the required SDC pump that is not in operation}	N/A	yes	no	N/A
4.9.9	3.9.6.1	The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs.	within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs	yes	yes	N/A
4.9.10	3.7.16.1	The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool.	at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
4.9.12.c	3.7.17.1	Verify at least once per 31 days the spent fuel pool boron concentration is greater than 2000 ppm.	at least once per 31 days	yes	yes	N/A
Section 3/4.10, Special Test Exceptions						
4.10.1.1	3.1.8.1	The position of each CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.	at least once per 2 hours	yes	yes	N/A
4.10.2.1	3.1.9.1	The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which any of the above requirements are suspended and shall be verified to be within the test power plateau.	at least once per hour during PHYSICS TESTS in which any of the above requirements are suspended	yes	yes	N/A
4.10.3.1	3.4.17.1	The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.	at least once per hour during startup and PHYSICS TESTS.	yes	yes	N/A
4.10.4.1	N/A	The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.	at least once per hour	no	yes	Millstone 2 St Lucie 1 and 2 Waterford 3
4.10.5.1	N/A	The Reactor Coolant System temperature and pressure relationship shall be verified to be within the acceptable region for operation of Figure 3.4-2 at least once per hour.	at least once per hour	no	yes	Millstone 2 St Lucie 1 and 2 Waterford 3
4.10.5.2	3.1.9.1	The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour.	at least once per hour	yes	yes	N/A

ANO-2 SR #	NUREG-1432 R4 SR #	ANO-2 Surveillance Description (NUREG 1432 Description if no ANO-2 Surveillance)	ANO-2 Surveillance Frequency	SR Frequency modified by TSTF-425	SR Frequency Modified in Amendment Request	Precedent with other Non-NUREG 1432 CE Plants
Section 3/4.11, Radioactive Effluents						
4.11.1	N/A	The quantity of radioactive material contained in each unprotected outside temporary radioactive liquid storage tank shall be determined to be within the above limit by analyzing a representative sample of the contents of the tank at least once per 7 days when radioactive materials are being added to the tank.	at least once per 7 days when radioactive materials are being added to the tank	no	yes	St Lucie 1 and 2 Waterford 3
4.11.2	N/A	The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank and the reactor coolant activity exceeds the limits of Specification 3.4.8.	at least once per 24 hours when radioactive materials are being added to the tank and the reactor coolant activity exceeds the limits of Specification 3.4.	no	yes	St Lucie 1 and 2
Section 6.5, Programs and Manuals						
6.5.2.b	5.5.2.b	Integrated leak test requirements for each system at least once per 18 months.	at least once per 18 months	no	yes	N/A
6.5.12.d	5.5.18.d (TSTF-425 SR 3.7.11.4)	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 VFTP, at a Frequency of one train every 18 months.	one train every 18 months	yes	yes	Waterford 3
6.5.13.c	5.5.13.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;	every 31 days	no	yes	N/A
6.5.18	5.5.20	N/A	N/A	yes	yes	N/A