Dominion Energy South Carolina, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060
DominionEnergy.com

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Dominion Energy

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

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Docket No.:	50-395
License No.:	NPF-12

DOMINION ENERGY SOUTH CAROLINA (DESC)
VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) UNIT 1

APPLICATION FOR TECHNICAL SPECIFICATION CHANGE REGARDING RISK-INFORMED JUSTIFICATION FOR THE RELOCATION OF SPECIFIC SURVEILLANCE FREQUENCY REQUIREMENTS TO A LICENSEE-CONTROLLED PROGRAM

In accordance with the provisions of Title 10 of the Code of Federal Regulations (10 CFR Part 50.90), "Application for Amendment of License, Construction Permit, or Early Site Permit," Dominion Energy South Carolina (DESC) hereby is submitting a request for an amendment to the Technical Specifications (TS) for Virgil C. Summer Nuclear Station (VCSNS) Unit 1.

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

Attachment 1 provides a description of the proposed change, the requested confirmation of applicability, and plant-specific variations. Attachment 2 provides documentation of PRA acceptability. Attachment 3 provides the existing TS pages marked-up to show the proposed change. Attachment 4 provides the revised (clean) TS pages. Attachment 5 provides the proposed TS Bases changes. Attachment 6 provides the No Significant Hazards Consideration. Attachment 7 provides a cross-reference of the proposed VCSNS TS changes with TSTF-425. Attachment 8 provides a list of references.

DESC requests approval of the proposed license amendment by October 9, 2022, with a 90-day implementation period.

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 South Carolina State Official.

Should you have any questions, please contact Mr. Yan Gao at (804)-273-2768.

Respectfully,

Mark D. Sartain
Vice President – Nuclear Engineering and Fleet Support

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mark D. Sartain, who is Vice President – Nuclear Engineering and Fleet Support of Dominion Energy South Carolina, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 8th day of April, 2021.

My Commission Expires: 12/31/24

CRAIG D SLY
Notary Public
Commonwealth of Virginia
Reg. # 7518653
My Commission Expires December 31, 2024

Notary Public

Attachments:

1. Description and Assessment
2. Documentation of PRA Acceptability
3. Existing TS Pages Mark-Up
4. Revised (Clean) TS Pages
5. Proposed TS Bases Changes
6. Proposed No Significant Hazards Consideration
7. Proposed VCSNS TS Changes Versus TSTF-425 Cross-Reference
8. References

cc: U.S. Nuclear Regulatory Commission, Region II
Marquis One Tower
245 Peachtree Center Avenue, NE
Suite 1200
Atlanta, Georgia 30303-1257

Mr. Vaughn Thomas
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 04 F-12
11555 Rockville Pike
Rockville, Maryland 20852-2738

NRC Senior Resident Inspector
V.C. Summer Nuclear Station

Ms. Anuradha Nair-Gimmi
Bureau of Environmental Health Services
South Carolina Department of Health and Environmental Control
2600 Bull Street
Columbia, SC 29201

Mr. G. J. Lindamood
Santee Cooper – Nuclear Coordinator
1 Riverwood Drive
Moncks Corner, SC 29461

ATTACHMENT 1

DESCRIPTION AND ASSESSMENT

Virgil C. Summer Nuclear Station (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

DESCRIPTION AND ASSESSMENT

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DESCRIPTION AND CHANGES

1.0 DESCRIPTION

The proposed amendment is to modify Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) [8.1] by relocating specific surveillance frequencies specified in the VCSNS TS to a licensee-controlled program [8.2] 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" [8.3]. Additionally, the change would add a new program, the Surveillance Frequency Control Program, to VCSNS TS [8.1] Section 6, Administrative Controls.

The changes are consistent with Nuclear Regulatory Commission (NRC) approved Technical Specification Task Force (TSTF) traveler TSTF-425, Revision 3 (ADAMS Accession No. ML090850642) [8.3]. The Federal Register notice published on July 6, 2009 [8.4] announced the availability of this TS improvement.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Dominion Energy South Carolina (DESC) has reviewed the safety evaluation dated July 6, 2009 [8.4]. This review included a review of the NRC staff's evaluation [8.4], TSTF-425, Revision 3 [8.3], and the requirements specified in NEI 04-10, Rev.1 [8.5].

Attachment 2 of this amendment request submittal includes DESC documentation with regard to PRA technical acceptability consistent with the requirements of Regulatory Guide 1.200 [8.6], Revision 2, Section 4.2, including documentation of the quality characteristics of those models in accordance with Regulatory Guide 1.200 [8.6].

DESC has concluded that the justifications presented in the TSTF proposal and the safety evaluation prepared by the NRC staff are applicable to VCSNS Unit 1 and justify this amendment to incorporate the changes to the VCSNS TS [8.1].

2.2 Optional Changes and Variations

The proposed amendment is consistent with Standard Technical Specifications (STS) changes described in TSTF-425, Revision 3 [8.3]. However, DESC is proposing the following administrative variations from TSTF-425 as discussed in section 2.2.1 through 2.2.3.

2.2.1 Administrative Changes/Variations

The proposed amendment is consistent with the Standard TS changes described in TSTF-425, Revision 3 [8.3]; however, DESC is proposing the following administrative variations from TSTF-425 with no impact on the NRC model safety evaluation dated July 6, 2009 [8.4].

1. The definition of STAGGERED TEST BASIS is being retained in VCSNS TS [8.1] section 1.0 Definitions since this terminology is used in TS [8.1], Procedures and Programs, Section 6.8.4.m, "Control Room Envelope Habitability Program," which is not the subject of this amendment request and is not proposed to be changed.

   VCSNS TS has test scheduling strategies for logic trains, channels and other components within systems that are also being relocated, consistent with the guidance of NEI 04-10, Rev. 1 [8.5]. Revision 1 to NEI 04-10 is provided to address test strategy (e.g. STAGGERED TEST BASIS) in addition to frequency. Under the proposed change, the Frequencies of all Surveillance Requirements (except those that reference other programs for the specific interval or that are event driven) are relocated. Similar to a STAGGERED TEST BASIS requirement, these VCSNS SRs require at least one logic train, channel or component to be tested within one interval, and all logic trains, channels or components to be tested with N intervals, where N is the total number of logic trains, channels or components subject to the test requirement. The following Surveillance Requirements (SRs) contain test scheduling requirements proposed for relocation:

   - SR 4.3.1.2, Reactor Trip System Instrumentation Response Time
   - SR 4.3.2.2, Engineered Safety Features Response Time

2. VCSNS TS 4.8.1.1.2.i.1 and 4.8.1.1.2.i.2 were renumbered as TS 4.8.1.1.2.i and 4.8.1.1.2.j respectively to allow separate surveillance frequencies.

3. NRC letter dated April 14, 2010 [8.7] provides a change to an optioned insert (INSERT #2) to the existing TS Bases to facilitate adoption of the traveler. The TSTF-425 TS Bases insert states the following:

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

   The above statement only applies to frequencies that have been changed in accordance with the Surveillance Frequency Control Program (SFCP) and does not apply to frequencies that are relocated but not changed. DESC has replaced the TSTF-425 TS Bases Insert #2 with the following:

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

4. Table 1.2, "Frequency Notation", of VCSNS TS, was revised to include the definition for "SFCP". SFCP is defined as "In accordance with the Surveillance Frequency Control Program."

5. Section 4.6.2.1.d of VCSNS TS is not being changed even though it is eligible. VCSNS may request a change to both the frequency and the method of testing in Section 4.6.2.1.d from at least once per 10 years to "following activities that could cause nozzle blockage" in a future License Amendment Request.

2.2.2 Technical Changes/Variations

VCSNS TS [8.1] varies significantly, especially in terms of format, from the Standard Technical Specifications as described in NUREG-1431 [8.30] and in TSTF-425 [8.3]. There are Surveillance Requirements (SRs) contained in TSTF-425 but not applicable or not contained in VCSNS TS. There are also SRs specific to (contained in) the VCSNS TS but not in NUREG-1431 or in TSTF-425. For some of these SRs specific to VCSNS TS, DESC has determined that the relocation of the associated SR frequencies is consistent with the intent of TSTF-425, Revision 3 and with the NRC's model safety evaluation dated July 6, 2009 (74 FR 31996) [8.4] including the scope exclusions identified in Section 1.0, "Introduction" of the model safety evaluation. The subject VCSNS TS specific SRs involve fixed periodic frequencies. 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 would be evaluated using the NRC approved methodology and probabilistic risk guidelines contained in NEI 04-10, Revision 1 [8.5].

Attachment 7 provides a cross-reference between the proposed changes in VCSNS TS [8.1] versus corresponding NUREG-1431 SRs and the corresponding SR location in TSTF-425 [8.3]. The first part (Part 1) of the Table in Attachment 7 provides comparisons only on the VCSNS TS SR frequencies proposed to be relocated, and the second part (Part 2) of the Table in Attachment 7 provides a list of the SRs included in NUREG-1431 but not applicable to VCSNS TS.

2.2.3 TS Bases Changes/Variations

Attachment 5 lists changes to the TS Bases to reflect the proposed TS changes shown in Attachment 3 and Attachment 4.

3.0 REGULATORY ANALYSIS

3.1 Applicable Regulatory Requirements and Criteria

A description of the proposed changes and their relationship to applicable regulatory requirements is provided in TSTF-425, Revision 3 [8.3] and the NRC's model safety evaluation published in the Notice of Availability dated July 6, 2009 (74 FR 31996) [8.4]. DESC has concluded that the relationship of the proposed changes to the applicable regulatory requirements presented in the Federal Register notice is applicable to VCSNS.

3.2 Precedent

This license amendment request application is being made in accordance with TSTF-425, Revision 3 [8.3]. DESC is not proposing significant variations or deviations from the TS changes described in TSTF-425 or in the content of the NRC staff's Model Safety Evaluation published on July 6, 2009 [8.4]. The NRC has previously approved license amendments to the TS to adopt TSTF-425. Some of the examples are Watts Bar Nuclear Plant, Unit 1 and 2, dated February 28, 2020 (ML20028F733), Palisades Nuclear Plant, dated December 30, 2019 (ML19317D855), Grand Gulf Nuclear Station, dated June 11, 2019 (ML19094A799), Arkansas Nuclear One Unit 1, dated May 22, 2019 (ML19098A955), River Bend Nuclear Plant, dated April 29, 2019 (ML19066A008), Donald C. Cook Nuclear Plant, Unit 1 and Unit 2, dated March 31, 2017 (ML17045A150), Shearon Harris Nuclear Power Plant, Unit 1, dated November 29, 2016 (ML16200A285) and R. E. Ginna Nuclear Power Plant, dated June 28, 2016 (ML16125A485).

VCSNS TSs are based on NUREG-0452 [8.34]. NRC approved a similar LAR submittal by Millstone Unit 3 [8.9], which has TSs that are also based on NUREG-0452.

This application proposes to relocate periodic surveillance frequencies to a licensee-controlled program and add a new program (the Surveillance Frequency Control Program) to the Administrative Controls section of VCSNS TS in accordance with TSTF-425 and as discussed in the previously approved amendments.

3.3 No Significant Hazards Consideration

DESC has reviewed the proposed no significant hazards consideration determination (NSHC) published in the Federal Register on July 6, 2009 (74 FR 31996) [8.4]. DESC has concluded that the proposed NSHC presented in the Federal Register notice is applicable to VCSNS Unit 1 and is provided as an Attachment (Attachment 6) to this amendment request which satisfies the requirement of 10 CFR 50.91(a) [8.8].

3.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in

the proposed manner; (2) such activities will be conducted in compliance with the NRC'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

DESC has reviewed the environmental consideration included in the model safety evaluation dated July 6, 2009 [8.4]. DESC has concluded that the staff's findings presented in the published evaluation are applicable to VCSNS and the evaluation is hereby incorporated by reference for this application.

ATTACHMENT 2

DOCUMENTATION OF PRA ACCEPTABILITY

DOCUMENTATION OF PRA ACCEPTABILITY

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DOCUMENTATION OF PRA ACCEPTABILITY

1.0 OVERVIEW

The implementation of the Surveillance Frequency Control Program (SFCP), also referred to as Technical Specifications Initiative 5b at VCSNS Unit 1 will follow the guidance provided in NEI 04-10, Revision 1 [8.5] in evaluating proposed surveillance test interval (STI; also referred to as "surveillance frequency") changes. The following steps of the risk informed STI revision process are common to all proposed STI changes within the proposed licensee-controlled program.

- Each proposed STI revision is reviewed to determine whether there are any commitments made to the Nuclear Regulatory Commission (NRC) that may prohibit changing the interval. If there are no related commitments, or the commitments may be changed using a commitment change process based on NRC endorsed guidance, then evaluation of the STI revision can proceed. If a commitment exists and the commitment change process does not permit the change without NRC approval, then the STI revision cannot be implemented. Only after receiving formal NRC approval to change the commitment could a STI revision proceed.
- A qualitative analysis is performed for each STI revision that involves several considerations as explained in NEI 04-10, Revision 1 [8.5].
- Each STI revision is reviewed by an expert panel, referred to as the Integrated Decision-making Panel (IDP), which is normally the same panel as is used for Maintenance Rule implementation, but with the addition of specialists with experience in surveillance tests and system or component reliability. If the IDP approves the STI revision, the change is documented and implemented, and available for future audits by the NRC. If the IDP does not approve the STI revision, the STI value is left unchanged.
- Performance monitoring is conducted as recommended by the IDP. In some cases, no additional monitoring may be necessary beyond that already conducted under the Maintenance Rule. Performance monitoring helps to confirm that any failure mechanisms significant to alter the basis provided in the justification for the surveillance interval change are subsequently identified.
- The IDP is responsible for periodic review of performance monitoring results. If it is determined that the time interval between successive performances of a surveillance test is a factor in the unsatisfactory outcome of the surveillance, the IDP returns the STI back to the previously acceptable STI.
- In addition to the above steps, the Probabilistic Risk Assessment (PRA) is used, when possible, to quantify the effect of a proposed individual STI revision compared to acceptance criteria in NEI 04-10, Revision 1 [8.5]. Also, the cumulative impact of risk informed STI revisions on PRA evaluations (i.e., internal and external events and

shutdown) is also compared to the risk acceptance criteria as delineated in NEI 04-10, Revision 1 [8.5].

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 STI revision.

The NEI 04-10, Revision 1 methodology endorses the guidance provided in Regulatory Guide (RG) 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." This regulatory guide has since been updated to RG 1.200, Revision 3 [8.33]. The guidance in RG 1.200, Revision 3 indicates that the following items should be defined by the licensee in order to demonstrate the Acceptability of a PRA used to support a risk-informed application:

1. Identify the scope of risk contributors addressed by the PRA model.
    - If not full scope (i.e., internal events, external events, applicable modes), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the PRA model.
2. Identify the parts of the PRA used to support the application.
    - PRA Model elements affected by the application and how these are implemented in the PRA model, including the cause-effect relationship as to how these elements would be impacted by the application.
3. Demonstrate the acceptability of the PRA.
    - Demonstrate that the model is up to date in that it represents the current plant design and configuration and represents current operating practices to the extent required to support the application.
    - Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by RG 1.200, Revision 3 [8.33]. Provide justification to show that, where specific requirements in the standard are not adequately met (including NRC staff clarification and qualification discussed in RG 1.200), it will not unduly impact the results.
    - Identify key assumptions and approximations relevant to the results used in the decision-making process.

2.0 PRA MODEL SCOPE

Surveillance Frequency Control is a broad risk informed application which potentially affects many Structures, Systems and Components (SSCs) at VCSNS (e.g., front-line, support safety related systems) modeled in the plant specific PRA models. All parts of the PRA models identified in the following subsections are applicable to the surveillance frequency control process. The methodology for using the PRA model to support risk-informed Technical Specifications frequencies is described in NEI 04-10 [8.5].

2.1 Internal Events/Flooding

The VCSNS Internal Events and Internal Flooding PRA model, known internally at DESC as VCS-R09 [8.10], has been revised recently by DESC in order to facilitate Surveillance Frequency Control as well as other applications of the PRA. This model revision addressed or partially addressed many open technical issues and findings, including many of the most technically complex and consequential issues. NRC has previously considered many of the internal events findings against this model in the context of Near-Term Task Force (NTTF) Recommendation 2.1, "Seismic" and NFPA 805 applications, Accession Nos ML19199A696 [8.11] and ML19305A005 [8.12], respectively.

The Internal Events/Flooding CDF/LERF generated by the VCS-R09 PRA model are:

CDF (/reactor-year)	LERF (/reactor-year)
2.7E-06	4.3E-07

2.2 Internal Fire

The VCSNS Fire PRA model, known internally at DESC as VCS-8ca [8.13], was developed by DESC (formerly SCE&G) to support implementation of its risk-informed, performance-based fire protection program in accordance with paragraph 50.48(c) of Title 10 of the *Code of Federal Regulations* (10 CFR) [*National Fire Protection Association Standard 805*, "Performance-Based Standard for Fire Protection for Light Water Reactor (LWR) Electric Generating Plants," (NFPA 805)]. NRC has previously reviewed the acceptability of this PRA model for NFPA 805 as discussed in the final Safety Evaluation for NFPA 805 for VCSNS [8.12]. No maintenance or update of the FPRA has been performed since this evaluation of the FPRA by NRC.

The Fire CDF/LERF generated by the VCS-8ca FPRA model are:

CDF (/reactor-year)	LERF (/reactor-year)
5.1E-05	2.7E-06

2.3 Seismic

The VCSNS seismic PRA model, known internally at DESC as VCS-8b [8.13], was generated in response to Near-Term Task Force (NTTF) Recommendation 2.1, "Seismic". NRC has previously evaluated the acceptability of this model in the context of the NTTF application, documented in ML19199A696 [8.11]. Following this NRC review, maintenance of the Seismic PRA model has been applied by DESC as discussed in a letter from DESC to USNRC, Serial No. 19-339B, dated October 9, 2019 [8.14]. This included removing credit for the RCP N9000 Abeyance Seal, and a more detailed human reliability analysis for emergency diesel generator (EDG) recovery following seismic induced relay chatter.

The Seismic CDF/LERF generated by the VCS-8b Seismic PRA model are:

CDF (/reactor-year)	LERF (/reactor-year)
3.5E-05	3.3E-06

2.4 PRA Summary

CDF/LERF at VCSNS from all PRA-modeled hazard groups are summarized in the table below:

Category	CDF (/reactor-year)	LERF (/reactor-year)
VCS-R09 Internal Events/Floods	2.7E-06	4.3E-07
VCS-8ca Fire	5.1E-05	2.7E-06
VCS-8b Seismic	3.5E-05	3.3E-06
Total	8.9E-05	6.4E-06

2.5 Other External Events

SCE&G submitted an IPEEE in June 1995 in response to Supplement 4 of Generic Letter 88-20. SCE&G did not identify any fundamental weaknesses or vulnerabilities to severe accident risk regarding other external events. The VCSNS hurricane, tornado and high winds analyses show that the plant is adequately designed, or procedures exist to cope with the effects of these natural events. Additionally, the VCSNS IPEEE demonstrated that transportation and nearby facility accidents were not considered to be significant vulnerabilities at the plant. In a letter dated June 14, 2000 [8.15], the NRC staff concluded that the submittal met the intent of Supplement 4 to Generic Letter 88-20, and that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities. Consistent with NEI 04-10, insights from this evaluation will be used to qualitatively analyze the risk impacts of these hazards associated with changes to STI.

2.6 Shutdown Risk Considerations

DESC does not maintain a shutdown PRA model for VCSNS. Consistent with the NEI 04-10, Revision 1 [8.5] guidance, qualitative information must be developed that supports the acceptability of the STI change with respect to the shutdown risk or it must be screened as not having an impact on the CDF and large early release frequency (LERF) metrics.

DESC operates under a shutdown risk management program to support implementation of NUMARC 91-06. The shutdown risk management implementing procedure SSP-004 provides guidelines for outage risk management which focuses on proper planning, conservative decision-making, maintaining defense in depth, and managing key safety functions. DESC will use the shutdown risk management program procedures to assess the potential impact on shutdown risk for proposed STI extensions, consistent with the guidance in NEI 04-10. Qualitative information will be developed that supports the

acceptability of the STI revision with respect to shutdown risk unless it can be screened as not having an impact on CDF and LERF.

3.0 ACCEPTABILITY OF VCSNS PRA MODEL

DESC employs a structured approach to establishing and maintaining the acceptability, adequacy and fidelity of the PRA models for all operating Dominion nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, supplemented with independent peer reviews. The following information describes this approach as it applies to the VCSNS PRA.

3.1 PRA Maintenance and Update

The VCSNS PRA model and documentation have been maintained as a living program. The PRA is routinely updated approximately every 3 to 5 years in order to reflect the current plant configuration and to reflect the accumulation of additional plant operating history and component failure data.

There are several procedures and GARDs (Guidance and Reference Documents) that govern DESC's PRA program. Procedure NF-AA-PRA-101 controls the maintenance and use of the PRA documentation and the associated NF-AA-PRA procedures and GARDs. These documents define the process to delineate the types of calculations to be performed, the computer codes and models used, and the method (or technique) by which each calculation is performed.

The NF-AA-PRA series of GARDs and procedures provide a detailed description of the methodology necessary to:

- Perform PRA for any station in the Dominion Nuclear Fleet, including VCSNS.
- Create and maintain products to support licensing and plant operation concerns for the Dominion Nuclear Fleet.
- Provide PRA model configuration control.
- Create and maintain configuration risk evaluation tools for the Dominion Nuclear Fleet.

A procedurally controlled process is used to maintain configuration control of the VCSNS PRA models, data, and software. In addition to model control, administrative mechanisms are in place to assure that plant modifications, procedure changes, operator training, system operation changes, and industry operating experiences (OEs) are appropriately screened, dispositioned, and scheduled for incorporation into the model. These processes help assure that the VCSNS PRA reflects the as-built, as-operated plant within the limitations of the PRA methodology.

This process involves a periodic review and update cycle to model any changes in the plant design or operation. Plant modifications and procedure changes are reviewed on an approximate quarterly or more frequent basis to determine if they impact the PRA and

if a PRA modeling and/or documentation change is warranted. These reviews are documented, and if any PRA changes are warranted, they are added to the PRA Configuration Control (PRACC) database for PRA implementation tracking.

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

3.2 Key PRA Assumptions and Sources of Uncertainty

The Technical Specifications Initiative 5b process is a risk-informed process with the PRA model results providing one of the inputs to determine if an STI change is acceptable. The NEI 04-10 methodology recognizes that a key area of uncertainty for this application is the 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.

Other relevant key assumptions will depend on the equipment being considered for an STI revision. Therefore, for each STI considered for revision, all PRA assumptions will be reviewed to identify potential key assumptions to the evaluation. These key assumptions will be evaluated in accordance with NEI 04-10 [8.5], Steps 5 and 14 to ensure there is no undue reliance on key assumptions in the Surveillance Frequency Control Process

3.3 PRA Upgrades

All changes applied to the VCSNS PRA have been categorized as PRA maintenance or have been peer reviewed as described in the following sections.

3.4 Peer Reviews

3.4.1 Internal Events PRA Peer Reviews

In March 2016, SCE&G requested that the PWROG support a full-scope peer review of the VCSNS Internal Events and Internal Flood PRA and review their Configuration Control Program to maintain, update, and upgrade the PRA [8.17]. This peer review was conducted to determine compliance with the ASME/ANS PRA Standard RA-Sa-2009 [8.16] for Internal Events, ASME/ANS PRA Standard RA-Sb-2013 [8.18] for Internal Flood, and RG 1.200 Revision 2 [8.6].

DESC completed an update of the internal events model in July 2020 which addressed or partially addressed some of the findings, including many of the most technically complex and consequential findings. Following that update the Internal Events model underwent a formal finding closure process as described in Appendix X to NEI 05-04

[8.19]/07-12 [8.20]/12-13 [8.21] which addressed the findings that were resolved in this update. In addition, a focused scope peer review [8.35] of PRA upgrades was performed, and new findings related to the focused scope were issued by the peer review team.

Sixty-five (65) findings remain open and active against the internal events PRA model. DESC has reviewed and prepared a disposition for all the findings with respect to potential impact on the Surveillance Frequency Control Program. These dispositions have been categorized as 'significant impact', 'minor impact' or 'no impact' to the surveillance frequency control program and are listed in the Tables in Section 4.1. Findings categorized as 'significant impact' are considered to have high likelihood to require detailed evaluation on a surveillance frequency specific basis in accordance with this process. Issues categorized as 'minor impact' are considered likely to be screened on a surveillance frequency specific basis with qualitative or bounding analysis during surveillance frequency evaluations. During the surveillance frequency evaluation process, if a non-trivial impact is expected due to any of these findings remaining open, then performance of additional sensitivity studies or PRA model changes to confirm the impact on the risk analysis will be included in the evaluation in accordance with NEI 04-10 [8.5] Steps 5 and 14 and DESC's PRA maintenance and upgrade process.

Even when a finding is considered completely resolved by DESC, the item is not considered closed and removed from tracking until it undergoes a formal closure process as described in Appendix X to NEI 05-04 [8.19], NEI 07-12 [8.20] and NEI 12-13 [8.21]. Therefore, issues considered to be completely resolved are still listed in the tables under Section 4.1. These resolved issues are considered open until formally closed out. DESC will continue to pursue resolution and closure of outstanding issues in accordance with its PRA maintenance and upgrade process.

The 2016 Internal Flood Peer Review was performed against the requirements of ASME/ANS PRA Standard RA-Sb-2013, which is a version of the PRA Standard that has not been formally endorsed by NRC in RG 1.200 or elsewhere. Therefore, DESC performed a gap assessment to determine if there were any aspects of the Internal Flood PRA which do not meet ASME/ANS RA-Sa-2009 and RG 1.200 because the PRA was reviewed against a non-endorsed PRA standard.

This review concluded that all the requirements of ASME/ANS RA-Sb-2013 fell into one of the following categories:

- Equivalent to ASME/ANS RA-Sa-2009 and RG 1.200, Revision 2, sometimes featuring a more explicit definition of the intent of the requirement
- More extensive than ASME/ANS RA-Sa-2009 and RG 1.200, Revision 2, such as requiring SSC susceptibility to HELB be identified
- Not Applicable to the VCSNS PRA
    - Specifically, IFEV-A8 quantitative screening is allowed in RA-Sb-2013 when a flood impacts multiple systems, but this is prohibited in RA-Sa-2009. However, the VCSNS PRA did not apply any quantitative screening and the peer review assessed IFEV-A8 as Not Applicable to VCSNS.

Therefore, DESC concluded that no additional reviews are needed to determine compliance with the ASME/ANS PRA Standard RA-Sa-2009 and RG 1.200, Revision 2 for the internal flood PRA.

3.4.2 Fire PRA Peer Reviews

A Fire PRA Peer Review was performed in December 2010 [8.22] against ASME/ANS PRA Standard RA-Sa-2009 [8.16] and RG 1.200, Revision 2 [8.6]. A follow-on Fire PRA Peer Review was performed in July 2011 [8.23], also against RA-Sa-2009 [8.16] and RG 1.200, Revision 2 [8.6]. These two peer reviews were focused scope, but when the two are combined they encompass all supporting requirements (SRs) in part 4 of RA-Sa-2009 (Fire) [8.16].

As part of the transition to NFPA 805 at VCSNS, DESC (formerly SCE&G) worked extensively to update and resolve open technical issues in the FPRA. All findings against the FPRA have been dispositioned to the extent necessary to conclude that the Fire PRA is technically adequate to demonstrate that the change in risk due to the transition to NFPA 805 meets the acceptance guidelines in RG 1.174 [8.24]. These dispositions are considered by DESC to be adequate to support the Surveillance Frequency Control Process at VCSNS in accordance with NEI 04-10 [8.5]. NRC has previously reviewed the acceptability of the Fire PRA with respect to the NFPA 805 application. NRC review is documented in the following locations:

- NFPA 805 Transition Safety Evaluation - ML14287A289 [8.25]
- FPRA Record of Review (ROR) - ML14311A128 [8.26]
- NFPA 805 Final Safety Evaluation - ML19305A005 [8.12]

Additionally, in 2020 all of the findings against the Fire PRA model underwent a formal finding closure process as described in Appendix X to NEI 05-04[8.19]/07-12[8.20]/12-13[8.21]. Seven (7) findings against the Fire PRA model remain open and four (4) SRs were considered NOT MET following this assessment. DESC considered the comments provided by the independent assessment team as to why these items were not closed and prepared a disposition for all the findings with respect to potential impact on the Surveillance Frequency Control Program. These dispositions are listed in the Tables under Section 4.2. During the surveillance frequency evaluation process, if a non-trivial impact is expected due to any of these findings remaining open, then the performance of additional sensitivity studies or PRA model changes to confirm the impact on the risk analysis will be included in the evaluation in accordance with NEI 04-10 Steps 5 and 14 and DESC's PRA maintenance and upgrade process.

3.4.3 Seismic PRA Peer Reviews

A peer review of the seismic hazard (SHA) element of the SPRA was conducted in July 2017 [8.27] against the supporting requirements of the ASME/ANS RA-Sa-2009 [8.16] along with clarifications from NRC Regulatory Guide 1.200, Revision 2 [8.6] [Note 1]. This

effort was part of the AP1000 SPRA review for units at the same site as Unit 1. This review was conducted using the peer review process defined in NEI 12-13 [8.21].

A peer review of the seismic fragility analysis (SFR) element and seismic plant response (SPR) element of the Unit 1 SPRA was performed by the Pressurized Water Reactor Owners Group in April 2018 [8.28] against the ASME/ANS RA-Sb-2013 [8.18] PRA Standard using the peer review process defined in NEI 12-13 [8.21].

In August 2018 an independent assessment was performed on the closure of the Findings and Observations (F&Os) previously generated against the VCSNS SPRA [8.29]. The independent assessment team used the process documented in Appendix X of NEI 05-04 [8.19], NEI 07-12 [8.20] and NEI 12-13 [8.21].

Of the 34 F&Os reviewed, the independent assessment team concurred that all except four (4) can be considered closed and all except two (2) supporting requirements (SRs) are considered MET to at least CC II. The Table in Section 4.3 lists the four open Findings in the Seismic PRA.

DESC has reviewed and prepared a disposition for all the findings for potential impact on the Surveillance Frequency Control Program. These dispositions are listed in the Table in Section 4.3. During the surveillance frequency evaluation process, if a non-trivial impact is expected from any of the findings, then performance of additional sensitivity studies or PRA model changes to confirm the impact on the risk analysis will be included in the evaluation in accordance with NEI 04-10 [8.5] Steps 5 and 14 and DESC's PRA maintenance and upgrade process.

Note 1: The SHA peer review also compared the SHA element to ASME/ANS RA-Sb-2013 [8.18] requirements. Where ASME/ANS RA-Sb-2013 [8.18] supporting requirements are more extensive than RA-Sa-2009 [8.16] and VCSNS SPRA conforms to RA-Sa-2009 [8.16] (and not necessarily RA-Sb-2013 [8.18]), this is noted in the supporting requirements assessments. Findings in this peer review were not written to require meeting RA-Sb-2013 [8.18] requirements if they are more restrictive than RA-Sa-2009 [8.16] and the SPRA meets RA-Sa-2009 [8.16].

3.4.4 Internal Events/Internal Flooding NOT MET or Capability Category I F&Os

ASME/ANS PRA standard contains a total of 323 numbered supporting requirements for internal events and internal flooding in nine technical elements. Of the 323 supporting requirements, 274 (84.8%) are rated as supporting requirements (SR) Met, Capability Category (CC) II or greater. Justifications of supporting requirements that did not have Capability Category II or greater will not unduly affect the Surveillance Frequency Control Program are included in the "SFCP Disposition" column in Section 4.1 through Section 4.3. Supporting requirements currently considered NOT MET or Capability Category I are summarized in the Table below.

CC I or NOT MET Supporting Requirements			
Supporting Requirement	**Assessment**	**Supporting Requirement**	**Assessment**
AS-A5	Not Met.	LE-C1	Not Met.
AS-A10	CC-I Met.	LE-C5	CC-I Met.
AS-B1	Not Met.	LE-C9	Not Met.
DA-A3	Not Met.	LE-C11	CC-I Met.
DA-C8	Not Met	LE-C12	CC-I Met.
DA-C10	Not Met	LE-D2	Not Met.
HR-B1	CC-I Met.	LE-D3	CC-I Met.
HR-H1	Not Met.	LE-D5	Not Met.
IE-A7	Not Met.	LE-G5	Not Met.
IE-A8	CC-I Met.	QU-D2	Not Met
IE-C2	Not Met.	SC-A5	Not Met.
IE-C6	Not Met.	SC-B3	Not Met.
IE-C9	Not Met.	SC-B4	Not Met.
IE-D1	Not Met.	SC-B5	Not Met.
IFEV-A1	CC-I Met.	SC-C1	Not Met.
IFEV-A6	CC-I Met.	SC-C2	Not Met.
IFEV-A7	Not Met.	SY-A4	Not Met.
IFQU-A1	Not Met.	SY-A11	Not Met.
IFSN-A1	Not Met.	SY-A13	Not Met.
IFSN-A3	Not Met.	SY-A14	Not Met.
IFSN-A4	Not Met.	SY-A15	Not Met.
IFSN-A6	Not Met.	SY-A18	Not Met.
IFSN-A8	Not Met.	SY-A22	Not Met.
		SY-B6	Not Met.
		SY-B12	Not Met.
		SY-B13	Not Met.

3.5 Summary

The VCSNS PRA technical capability evaluations and the maintenance and update processes described above provide a robust basis for concluding that the VCSNS PRA is suitable for use in risk-informed processes such as that proposed for the implementation of the Surveillance Frequency Control Program. In addition to the standard set of sensitivity studies required per the NEI 04-10, Revision 1 [8.5] methodology, changes at the facility, PRA upgrades, uncertainties and gaps to specific requirements in the ASME/ANS PRA standard will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

4.0 F&O TABLES

4.1 Internal Events PRA F&Os

Internal Events PRA F&Os and SFCP Dispositions			
Part 1 – F&Os with Significant Impact to SFCP			
FOID	**Description (Note: in this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
02-11	Loss of an AC bus, failure of the chilled water system, and consequential LOOP (post trip, after another initiating event occurs) are missing from the documentation or were screened using an inadequate basis. For example, in section 7.2.4.4, for loss of an AC bus the criteria listed does not discuss the impact of having to administratively shut the plant down within a short time frame due to LCO requirements with an entire electrical bus out of service. (This F&O originated from SR IE-C6.)	IE-C6	<u>**OPEN**</u>: Chilled water has been reviewed and screened from the PRA model. The impact of the remaining scope of this issue is relevant to surveillance evaluations related to loss of AC Power. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
04-01	The current model utilizes mission times less than 24 hours by the statement #1. (This F&O originated from SR SC-A5)	SC-A5	<u>**PARTIALLY RESOLVED**</u>: All reliability event data in the model has been re-calculated on a 24-hour mission time basis, but some accident sequences in the model need to be expanded to fully account for the 24hr mission time. The impact of this issue will be confirmed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
04-26	Secondary side isolation has not been modeled realistically for SGTR sequences since these were assumed not to result in LERF. (This F&O originated from SR LE-D5)	LE-D5	<u>**OPEN**</u>: The scope of the impact of this issue is limited to Core Damage and LERF sequences that involve failure of Secondary Isolation. The impact of this issue will be assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
05-27	Mitigating system dependencies as a result of an initiating events are not clearly described and as modeled will result in skewing of the importance. (This F&O originated from SR AS-B1)	AS-B1	<u>**OPEN**</u>: It is anticipated that resolving this issue has some impact on specific accident sequences within the VCSNS PRA. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
DA-C8-01	No attempt was made to estimate the time that components were configured in standby status as	DA-C8	<u>**OPEN**</u>: The impact of this issue will be carefully considered when

Internal Events PRA F&Os and SFCP Dispositions			
Part 1 – F&Os with Significant Impact to SFCP			
FOID	**Description (Note: In this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
	required for Category I. Section 3.3 of GARD 2061 discusses the need to use operational data to develop basic events used to represent the fraction of time a component is in standby and that Category II cannot be met by simply estimating a pump is in standby 50% of the time.		performing surveillance frequency evaluations in order to ensure that PRA assumptions regarding standby train alignment are not unduly biasing PRA insights and outcomes. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
DA-C16-01	The LOOP recovery data is about ten years old.	DA-C16	**OPEN**: The Scope of this issue is limited to LOOP Recovery Data. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.

Internal Events PRA F&Os and SFCP Dispositions			
Part 2 – F&Os with Minor Impact to SFCP			
FOID	**Description (Note: In this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
02-09	A review of recent LERs and plant initiator precursor events could not be found. The initiating events analysis is based on DC00300-150, Rev. 0, which only considered LERs through 3/22/08. (This F&O originated from SR IE-C2)	IE-C2	**OPEN**: This review is judged to be very unlikely to reveal new PRA insights, therefore impact of resolving this issue on the Surveillance Frequency Control Program is judged to be very small. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
02-13	ISLOCA analysis performed in CN-RAM-14-030 documented the applicable ISLOCA pathways based on the plant features and procedures in accordance with WCAP-17154-P; see Appendix B of CN-RAM-14-030. However, the treatment of ISLOCA pathway 410-S appears to be overly conservative. (This F&O originated from SR IE-C14.)	IE-C14	**OPEN**: Conservatively treating ISLOCA pathways applies conservative bias to risk calculations and is therefore acceptable for use in the Surveillance Frequency Control Application, which utilizes absolute risk metrics (i.e. delta CDF, delta LERF) as acceptance criteria rather than metrics which are relative to total CDF/LERF

Internal Events PRA F&Os and SFCP Dispositions			
Part 2 – F&Os with Minor Impact to SFCP			
FOID	**Description (Note: in this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
			such as RAW, Fussell-Vesely, etc. If it is determined that this approach causes a surveillance frequency evaluation to fail to meet its PRA acceptance criteria or is otherwise unable to generate meaningful results, then the PRA model will be reviewed and revised to include the level of detail needed to complete the evaluation in accordance with NEI 04-10.
02-18	Examples were found where the Time window for success is not based on AS/SC based timings, (e.g. OA_AAC_SBO, an assumption was used in lieu of an actual AS timing value which should be available. Additionally, for other HFE the limiting timing for which it is applied was not used, e.g. BCPM--XPP39CHE is used for both LOOP and LCCW scenarios but the limiting timing was not selected). (This F&O originated from SR HR-F2.)	HR-F2	<u>**PARTIALLY RESOLVED**</u>: This issue was independently reviewed for closure during the 2020 Appendix X closeout activity. The closeout team concluded that recent revisions to HRA analysis and documentation to use realistic AS/SC timing information (e.g., based on MAAP, operator interviews, industry analysis) are sufficient to resolve the Finding, but that for the specific examples identified in the F&O the documentation provided no discussion that the timing was based on the more limiting accident sequence. The impact of resolving the remainder of this issue on the Surveillance Frequency Control Program is judged to be small. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved
02-22	Some risk-significant operator recovery actions are not currently modeled in the internal events model. Such events should be included as they could significantly impact the results. Examples include an operator action to align recirculation for the RHR system and close the PORV block valve after the PORV has been challenged, or terminate SI after the SRV has been challenged, and to manually align equipment following a failure of ESFAS. Action OAESF1/OAESF2/OAESF3 were provided by VCS but did not appear in the modeled logic and should be included as it allows operators to start equipment after a failure as directed by procedures. (This F&O originated from SR HR-H1.)	HR-H1	<u>**OPEN**</u>: Not taking credit for operator recovery actions applies conservative bias to risk calculations and is therefore acceptable for use in the Surveillance Frequency Control Application, which utilizes absolute risk metrics (i.e. delta CDF, delta LERF) as acceptance criteria rather than metrics which are relative to total CDF/LERF such as RAW, Fussell-Vesely, etc. If it is determined that this approach causes a surveillance frequency evaluation to fail to

Internal Events PRA F&Os and SFCP Dispositions			
Part 2 – F&Os with Minor Impact to SFCP			
FOID	**Description (Note: in this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
			meet its PRA acceptance criteria or is otherwise unable to generate meaningful results, then the PRA model will be reviewed and revised to include the level of detail needed to complete the evaluation in accordance with NEI 04-10 Steps 9 & 11.
03-16	There is no evidence that additional demands from post-maintenance testing were considered. (This F&O originated from SR DA-C6)	DA-C6	**PARTIALLY RESOLVED**: Processes and documentation related to the data element of VCSNS PRA have been updated using Dominion fleet standard approaches. The model documentation now discusses post-maintenance testing in data sources. The impact of post maintenance testing on VCSNS Data analysis is small because only data collected using the plant computer includes post maintenance testing in the count, which is one of several data sources in the analysis. More detailed sensitivity analysis is required to completely resolve this issue. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
04-04	Thermal hydraulic analyses have been used to support the success criteria. However, many issues were identified with success criteria. For example, In an SGTR event with failure of SG isolation (SGI) and RWST Refill (RF), core damage is avoided if (long-term heat removal) LTHR is successful. It's not clear how success is achieved if all this function includes is steam relief. Or is it assumed that this function is similar to Cooldown function? Is there a MAAP analysis to support this success criteria? Another example, bleed and feed success criteria MAAP case assume the action is initiated at the time of the cue (12% wide range level) which allows no time for the operator action. With the extra time used before the action is actually initiated, it is not clear whether the case would still be successful with 1 Charging pump and 1 PORV. Yet another example, Assumption 3 in Section 5.0 states that 10 hours was used as a mission time for emergency feedwater in an SLOCA event since it is used for RCS cooldown and depressurization only if HPI fails. Looking at the	SC-B3	**OPEN**: A comprehensive review has been performed to identify outstanding gaps. The Success Criteria have been updated over time in several reference documents. These documents need to be more clearly organized and the success criteria directly referenced to in the analysis. It is anticipated that completely resolving this issue will have no impact on the Surveillance Frequency Control Program. This will be confirmed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.

Internal Events PRA F&Os and SFCP Dispositions			
Part 2 – F&Os with Minor Impact to SFCP			
FOID	**Description (Note: In this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
	SLOCA event tree and fault tree sequence SLO-7, there is no mitigation if HPI fails (core damage occurs). (This F&O originated from SR SC-B3)		
04-06	The only comparison performed was for a SGTR event, with no results shown. This is not enough to demonstrate reasonableness and acceptability of the results. (This F&O originated from SR SC-B5)	SC-B5	**OPEN**: It is anticipated that resolving this issue has a very small impact on the VCSNS PRA. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
04-11	The LE-A1 and LE-A2 characteristics are accounted for using different methods including treatment in Level 2 CETs, bridge trees, PDS. One finding was identified. (This F&O originated from SR LE-A4)	LE-A4	**OPEN**: Severe accident sequence binning in the VCSNS PRA has been performed in accordance with WCAP-16341-P "Simplified Level 2 Modeling Guidelines", which is in turn consistent with event tree analysis performed in NUREG-6595 with some additional details intended to satisfy CCII of the ASME/ANS RA-Sa–2009 Standard. WCAP-16341-P was created with the intent to be applicable to PWRs such as VCSNS, but no specific justification using plant specific severe accident analyses exists to justify this. It is anticipated that resolving this issue will have no impact on the Surveillance Frequency Control Program. This will be confirmed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
04-13	All LERF contributors from the table have been considered in LERF analysis. However, in-vessel injection was not credited. (This F&O originated from SR LE-B1)	LE-B1	**OPEN**: Not taking credit for in-vessel recovery to stop core-melt process applies conservative bias to risk calculations and is therefore acceptable for use in the Surveillance Frequency Control Application, which utilizes absolute risk metrics (i.e. delta CDF, delta LERF) as acceptance criteria rather than metrics which are relative to total CDF/LERF such as RAW, Fussell-Vesely, etc. If it is determined that a conservative accident sequence

Internal Events PRA F&Os and SFCP Dispositions			
Part 2 – F&Os with Minor Impact to SFCP			
FOID	**Description (Note: In this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
			modeling approach causes a surveillance frequency evaluation to fail to meet its PRA acceptance criteria or is otherwise unable to generate meaningful results, then the PRA model will be reviewed and revised to include the level of detail needed to complete the evaluation in accordance with NEI 04-10 Steps 9 & 11.
04-14	No severe accident analyses exist to support determination on whether or not an accident sequence should be binned to LERF. (This F&O originated from SR LE-C1)	LE-C1	**<u>OPEN</u>**: Severe accident sequence binning in the VCSNS PRA has been performed in accordance with WCAP-16341-P "Simplified Level 2 Modeling Guidelines", which is in turn consistent with event tree analysis performed in NUREG-6595 with some additional details intended to satisfy CCII of the ASME/ANS RA-Sa–2009 Standard. WCAP-16341-P was created with the intent to be applicable to PWRs such as VCS, but no specific justification using plant specific severe accident analyses exists to justify this. It is anticipated that resolving this issue will have no impact on the Surveillance Frequency Control Program. This will be confirmed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
04-16	Model logic is included for realistic estimation of accident progression sequences resulting in a LER. This is documented in notebook CN-RAM-14-035, Section 6.0 and Appendix A. A finding is identified. (This F&O originated from SR LE-C4)	LE-C4	**<u>OPEN</u>**: Severe Accident Progression and Operator actions within the scope of Westinghouse Severe Accident Management Guidance have been considered in the VCSNS PRA in accordance with WCAP-16341-P "Simplified Level 2 Modeling Guidelines", which is in turn consistent with event tree analysis performed in NUREG-6595 with some additional details intended to satisfy CCII of the ASME/ANS RA-Sa–2009 Standard. WCAP-16341-P was created with the intent to be applicable to PWRs such as VCSNS, but no specific justification using plant specific

Internal Events PRA F&Os and SFCP Dispositions			
Part 2 – F&Os with Minor Impact to SFCP			
FOID	**Description (Note: in this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
			severe accident analyses exists to justify this. It is anticipated that resolving this issue will have no impact on the Surveillance Frequency Control Program. This will be confirmed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
04-17	Realistic or plant specific analyses have not been performed for system success criteria for significant accident progression sequences that result in LER. (This F&O originated from SR LE-C5)	LE-C5	**OPEN**: Post Core Damage System Success Criteria have been considered in the VCSNS PRA in accordance with WCAP-16341-P Simplified Level 2 Modeling Guidelines, which is in turn consistent with event tree analysis performed in NUREG-6595 with some additional details intended to satisfy CCII of the ASME/ANS RA-Sa–2009 Standard. WCAP-16341-P was created with the intent to be applicable to PWRs such as VCSNS, but no specific justification using plant specific severe accident analyses exists to justify this. It is anticipated that resolving this issue will have no impact on the Surveillance Frequency Control Program. This will be confirmed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
04-20	Pressurizer PORVs have been credited post-core damage for RCS depressurization. No justification is provided for PORV survivability in harsh environment inside containment after core damage. (This F&O originated from SR LE-C9)	LE-C9	**OPEN**: In accordance with WCAP-16679-P PORV survivability post core damage is treated as a model assumption/uncertainty in the Level 2 analysis and consideration will be given to assess impact of this assumption on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14). This finding is considered open until model documentation is finalized.
04-21	It was assumed that early containment failures result in core damage. Therefore, credit for equipment survivability after containment failure has not been taken.	LE-C11	**OPEN**: Not taking credit for equipment survivability after containment failure applies conservative bias to risk

Internal Events PRA F&Os and SFCP Dispositions			
Part 2 – F&Os with Minor Impact to SFCP			
FOID	**Description (Note: In this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
	(This F&O originated from SR LE-C11)		calculations and is therefore acceptable for use in the Surveillance Frequency Control Application, which utilizes absolute risk metrics (i.e. delta CDF, delta LERF) as acceptance criteria rather than metrics which are relative to total CDF/LERF such as RAW, Fussell-Vesely, etc. If it is determined that a conservative accident sequence modeling approach causes a surveillance frequency evaluation to fail to meet its PRA acceptance criteria or is otherwise unable to generate meaningful results, then the PRA model will be reviewed and revised to include the level of detail needed to complete the evaluation in accordance with NEI 04-10 Steps 9 & 11.
04-25	This assessment is based on VC Summer self-assessment. Also, no analysis exists to determine if containment failure location has any impact on binning to LERF. (This F&O originated from SR LE-D3)	LE-D3	<u>**OPEN**</u>: Containment Failure location is treated conservatively in the VCSNS PRA (i.e. all containment failures are binned to LERF). This applies conservative bias to risk calculations and is therefore acceptable for use in the Surveillance Frequency Control Application, which utilizes absolute risk metrics (i.e. delta CDF, delta LERF) as acceptance criteria rather than metrics which are relative to total CDF/LERF such as RAW, Fussell-Vesely, etc. If it is determined that a conservative accident sequence modeling approach causes a surveillance frequency evaluation to fail to meet its PRA acceptance criteria or is otherwise unable to generate meaningful results, then the PRA model will be reviewed and revised to include the level of detail needed to complete the evaluation in accordance with NEI 04-10 Steps 9 & 11.
04-30	Plant specific assumptions are identified and documented in Section 5 of notebook CN-RAM-14-035, and generic uncertainties are characterized in Section 7.4. In addition, certain sensitivity cases were performed and documented in Section 7.3. LERF aleatory uncertainty evaluation is performed with	LE-F3	<u>**OPEN**</u>. It is anticipated that DESC's resolution of this finding will include adjustment and technical justification of existing uncertainty parameters in the Level 2 model. Surveillance

Internal Events PRA F&Os and SFCP Dispositions			
Part 2 – F&Os with Minor Impact to SFCP			
FOID	**Description (Note: in this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
	UNCERT in Section 7.2.2. One finding was identified on error factors. (This F&O originated from SR LE-F3)		frequency evaluations are considered generally to have low sensitivity to Level 2 uncertainty parameters. This will be confirmed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
05-02	The approach did identify a measurable change which was used to find an adjustment factor for SOKC. However, the analysis is believed to be incomplete. (This F&O originated from SR QU-A3)	QU-A2	<u>**OPEN**</u>: The impact of resolving this issue on the surveillance frequency control program is judged to be small, the scope of affected cutsets is limited to cutsets with event combinations that use the same typecode (coupled events) which is rare. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
05-05	Cutsets and sequences have been reviewed to determine whether they accurately reflect plant design and operation. However, based on the overly conservative assumptions that go into the constituent analyses, some of the dominant cutsets are judged not to be realistic. (This F&O originated from SR QU-D2)	QU-D2	<u>**OPEN**</u>: Exclusion of alternative mitigation capabilities from the PRA model applies conservative bias to risk calculations and is therefore acceptable for use in the Surveillance Frequency Control Application, which utilizes absolute risk metrics (i.e. delta CDF, delta LERF) as acceptance criteria rather than metrics which are relative to total CDF/LERF such as RAW, Fussell-Vesely, etc. If it is determined that a conservative accident sequence modeling approach causes a surveillance frequency evaluation to fail to meet its PRA acceptance criteria or is otherwise unable to generate meaningful results, then the PRA model will be reviewed and revised to include the level of detail needed to complete the evaluation in accordance with NEI 04-10 Steps 9 & 11.
05-21	**NOTE: The text of this finding has been abbreviated due to excessive length and technical detail describing potential conservatisms in accident sequence analysis. Full detail is available from DESC upon request.**	AS-A5	<u>**OPEN**</u>: Exclusion of alternative mitigation capabilities from the PRA model applies conservative bias to risk calculations and is therefore acceptable for use in

Internal Events PRA F&Os and SFCP Dispositions			
Part 2 – F&Os with Minor Impact to SFCP			
FOID	**Description (Note: in this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
	Several conservatisms were noted during review of the accident sequences where procedural pathways provide additional potential mitigating capabilities that could be credited and have been credited at other similar plants. As the exclusion of these mitigating pathways has a potential to impact the results of the model and importance measures, these sequences should be reviewed, and the additional mitigating capabilities should be incorporated or appropriately dispositioned: 1) Loss of EFW – Given a loss of EFW there is a potential to restore MFW or condensate prior to initiating feed and bleed. 2) HPI Requirement on SSB – The accident sequence and event tree currently require HPI in the event of an SSB regardless of if a bleed feed requirement arises. If the RCS pressure boundary is not breached HPI should not be required as RCS volume should remain sufficient such that core uncover does not occur. 3) RWST in Small LOCAs – Currently the capability to re-fill the RWST and remain on injection given a failure of recirculation is not modeled. RWST re-fill is currently only credited on a SGTR or an ISLOCA. 4) SLOCA where HPI fails - There is no credit for depressurization and low-pressure injection currently in the accident sequences. 5) Failure of Recirculation Auto-Swap: Review of cutsets revealed that the failure of relays (due to CCF) in the RHR recirculation valves fails the recirculation function however it does not appear as if an operator action to manually align for recirculation. (This F&O originated from SR AS-A5)		the Surveillance Frequency Control Application, which utilizes absolute risk metrics (i.e. delta CDF, delta LERF) as acceptance criteria rather than metrics which are relative to total CDF/LERF such as RAW, Fussell-Vesely, etc. If it is determined that a conservative accident sequence modeling approach causes a surveillance frequency evaluation to fail to meet its PRA acceptance criteria or is otherwise unable to generate meaningful results, then the PRA model will be reviewed and revised to include the level of detail needed to complete the evaluation in accordance with NEI 04-10 Steps 9 & 11.
06-04	The model identifies spray, flood and HELB considerations for piping considerations. Human induced and some other scenarios such as expansion joints were not considered. (This F&O originated from SR IFSO-A4)	IFSO-A4	**PARTIALLY RESOLVED:** The station equipment list has been reviewed and it has been determined that all expansion joints are located at the turbine building 412 ft elevation. Flood sources in this area are treated with a conservative (bounding) frequency because of the very low risk significance of flooding in this area. Flooding Frequency due to expansion joint failures is considered subsumed to the existing bounding flood

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Part 2 – F&Os with Minor Impact to SFCP			
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			frequencies. Resolution of this issue is expected to have no impact on the Surveillance Frequency Control Program. Refer to similar finding 06-28 for disposition of human-induced scenarios.
06-05	After a review of the characterization of the scenarios, it was found that the grouping was based on a zone and system but did not consider timing, component impacts based on different line flow rates which may lead to conservative results for risk-significant scenarios. (This F&O originated from SR IFEV-A1)	IFEV-A1	**OPEN**: Not differentiating impacts of flooding based on timing and break size applies conservative bias to risk calculations and is therefore acceptable for use in the Surveillance Frequency Control Application, which utilizes absolute risk metrics (i.e. delta CDF, delta LERF) as acceptance criteria rather than metrics which are relative to total CDF/LERF such as RAW, Fussell-Vesely, etc. If it is determined that this approach causes a surveillance frequency evaluation to fail to meet its PRA acceptance criteria or is otherwise unable to generate meaningful results, then the PRA model will be reviewed and revised to include the level of detail needed to complete the evaluation in accordance with NEI 04-10 Steps 9 & 11.
06-06	The initiating event frequency was based on generic information. (This F&O originated from SR IFEV-A6)	IFEV-A6	**OPEN**: Dominion Energy has proceduralized the exclusive use of generic data sources to determine flood initiating event frequencies in PRA models, unless an adverse flooding trend exists in the plant operating experience. It is anticipated that Dominion's resolution of this finding will include an operating experience review to confirm that no adverse trends exist. Surveillance frequency evaluations are considered generally to have very low sensitivity to Internal Flood frequencies. This will be confirmed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.

Internal Events PRA F&Os and SFCP Dispositions			
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06-09	No documentation of automatic features and operator responses could be found that could terminate or contain flood propagation. For example, the top cutset which is a BD line break in IB, does not have a failure of automatic isolation of the BD line. (This F&O originated from SR IFSN-A3)	IFSN-A3	**OPEN**: Not taking credit for automatic and operator responses to flood initiators applies conservative bias to risk calculations and is therefore acceptable for use in the Surveillance Frequency Control Application, which utilizes absolute risk metrics (i.e. delta CDF, delta LERF) as acceptance criteria rather than metrics which are relative to total CDF/LERF such as RAW, Fussell-Vesely, etc. If it is determined that this approach causes a surveillance frequency evaluation to fail to meet its PRA acceptance criteria or is otherwise unable to generate meaningful results, then the PRA model will be reviewed and revised to include the level of detail needed to complete the evaluation in accordance with NEI 04-10 Steps 9 & 11.
06-10	The capacity of the drains is estimated in Appendix A of notebook CN-RAM-13-046. However, no calculation could be found on water retention in sumps, curbs, dikes, etc. (This F&O originated from SR IFSN-A4)	IFSN-A4	**OPEN**: The current treatment of not including water retention in sumps, curbs, dikes, etc in flood timing and recovery calculations applies conservative bias to flood risk calculations and is therefore acceptable for use in the Surveillance Frequency Control Application, which utilizes absolute risk metrics (delta CDF, delta LERF) rather than metrics which are relative to total CDF/LERF such as RAW, Fussell-Vesely, etc. If it is determined that this approach causes a surveillance frequency evaluation to fail to meet its PRA acceptance criteria or is otherwise unable to generate meaningful results, then the PRA model will be reviewed and revised to include the level of detail needed to complete the evaluation in accordance with NEI 04-10 Steps 9 & 11.
06-13	PRA notebook CN-RAM-13-044 Section 7.4 states that drains and backflow have been considered in the flooding analysis. However, no evidence could be	IFSN-A8	**OPEN**: Drain backflow flood propagation pathways require review and disposition in the internal flood PRA. The

Internal Events PRA F&Os and SFCP Dispositions			
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	found on actually considering drains and backflow for inter-area flood propagation. (This F&O originated from SR IFSN-A8)		resolution of this issue is expected to have a small impact on the Internal Flood PRA because floods are already modeled as propagating to broad areas of the lower elevations of Auxiliary, Control, Diesel, Intermediate and Turbine Buildings (non-screened flood buildings) in a bounding propagation approach. The Impact of this issue will be confirmed on a surveillance evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
06-14	Flood rate calculations are performed using EPRI Tech Report formula, and compared with the system flow rates, and the lower is used in the timing calculation. This may be non-conservative since in some cases lower than expected flow rates may be used. Suggestion: Timing calculations are performed based on estimation of flood height in an 8-hr. time period, then back calculating the available time before PRA equipment fails in the area using the break flow rate. It is suggested to calculate the timing based on first estimating the critical flood volume and then dividing it by the break flow rate. (This F&O originated from SR IFSN-A9)	IFSN-A9	**<u>OPEN</u>**: The scope of this issue is limited to core damage and large early release sequences that involve catastrophic internal flooding events and PRA modeled flood isolation recoveries. The impact of this issue on the surveillance frequency control program is very small because these sequences generally do not require random component faults to progress to core damage. The impact of this issue will be assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
06-15	Impact of flooding on internal events HEPs was considered but it appears that the effect on HEPs was performed with not sufficient basis. For example, from notebook CN-RAM-13-048, "For Tdelay ≤ 10 minutes, an increase of two minutes was added to Texe and Tcog in the time window. For Tdelay > 10 minutes, an increase of five minutes was added to Texe and Tcog." This appears to have no technical basis. (This F&O originated from SR IFQU-A6)	IFQU-A6	**OPEN**: The impact of this issue on the surveillance frequency control program is judged to be small because internal flooding sequences are not typically significant risk contributors in surveillance frequency evaluations. The impact of this issue will be assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
06-18	The system models exclude in many cases component failures on the basis of guidance in PSA-01. For example, Locked open manual valves XVG-8471A, B, C and normally open manual valve XVG-	SY-A11	**<u>PARTIALLY RESOLVED</u>**: System model screening was re-reviewed to identify additional components that need to be added to the PRA model. These

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	8388 are not modeled. No quantitative considerations are specified to meet SY-A15. (This F&O originated from SR SY-A11)		components have been added to the PRA model. The outstanding resolution of the issue includes systematically incorporating system model screening into PRA model development process and completing model documentation. System model screening is judged to have a small impact on the surveillance frequency control program because risk insights are generally governed by major components in modeled systems. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
06-19	1/3 rule used and too many divergence paths not addressed. (This F&O originated from SR SY-A13)	SY-A13	**PARTIALLY RESOLVED**: System model screening was re-reviewed to identify additional components that need to be added to the PRA model. These components have been added to the PRA model. The outstanding resolution of the issue includes systematically incorporating system model screening into PRA model development process and completing model documentation. System model screening is judged to have a small impact on the surveillance frequency control program because risk insights are generally governed by major components in modeled systems. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
06-20	Overall modeling is considered consistent with the data notebook and typical generic databases. There are a few instances where the models are inconsistently applied across some of the notebooks due to the use of screening. Some other failure modes were excluded as failures although data existed in the data notebook and other generic sources. An example is the exclusion of air dryers as a failure mode. A suggestion is made that the models be reviewed for compliance to the standard and that all modeling	SY-A14	**PARTIALLY RESOLVED**: System model screening was re-reviewed to identify additional components that need to be added to the PRA model. These components have been added to the PRA model. The outstanding resolution of the issue includes systematically incorporating system model screening into PRA

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	assumptions be based on a single standard. Currently there are two referenced standards that may be in conflict (PSA-01 and CN-RAM-13-020). An example is the statement in the CC notebook: "Modeling note 18 below was obtained from Reference 22. Note that some modeling notes may require additional documentation or changes due to differences with the screening criteria developed in Reference 13." (This F&O originated from SR SY-A14)		model development process and completing model documentation. System model screening is judged to have a small impact on the surveillance frequency control program because risk insights are generally governed by major components in modeled systems. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
06-21	It appears that the original system fault tree screening was developed, or at least consolidated in PSA 01, (Attachments I/II) but was later updated in CN-RAM-13-020 as an assessment revealed that the PSA-01 screening criteria were insufficient to meet the standard. CN-RAM-13-020 also suggests that PSA-01 be updated with the new guidance. However, the CCW system NB (CN-RAM-14-022) lists both of these documents as references (References 13 and 21) while in contrast the EF Notebook only lists the updated document CN-RAM-13-020. Section 7.3.1 of the CCW NB says on page 36 that "some modeling notes may require additional documentation or changes due to differences with the screening criteria developed". It appears that the revised guidance was not utilized in a consistent assessment approach. (This F&O originated from SR SY-A15)	SY-A15	<u>**PARTIALLY RESOLVED**</u>: System model screening was re-reviewed to identify additional components that need to be added to the PRA model. These components have been added to the PRA model. The outstanding resolution of the issue includes systematically incorporating system model screening into PRA model development process and completing model documentation. System model screening is judged to have a small impact on the surveillance frequency control program because risk insights are generally governed by major components in modeled systems. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
06-24	Review of the system notebooks indicates a thorough analysis of each system that identifies support systems and varying operating conditions which may impact accident scenarios. HVAC not addressed for CSIP and RHR. There are no analyses or discussions provided for other rooms. (This F&O originated from SR SY-B6)	SY-B6	<u>**PARTIALLY RESOLVED**</u>: The justification for HVAC success criteria for charging/SI pump rooms and RHR/Spray Pump rooms was reviewed and it was determined that no changes to the PRA model are required. The PRA model systems analysis has been reviewed and adjusted to account for HVAC dependencies more thoroughly, but system modeling and documentation of these dependencies remain incomplete. This issue will be reviewed and assessed on an

Internal Events PRA F&Os and SFCP Dispositions			
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			evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
06-26	This should be coupled with the issues with SY-A15(a). In addition to showing the screened components do not impact the system there should also be an assessment that they do not impact multiple systems (i.e. many of the support systems that use screening). (This F&O originated from SR SY-B13)	SY-B13	**PARTIALLY RESOLVED**: The quantitative screening was re-reviewed to identify additional components that need to be added to the PRA model. These components have been added to the PRA model. The outstanding resolution of the issue includes systematically incorporating system model screening into PRA model development process and completing model documentation. System model screening is judged to have a small impact on the surveillance frequency control program because risk insights are generally governed by major components in modeled systems. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
DA-A3-01	When determining pump start failures, the time-dependent failure data in the first hour of run time was improperly combined with demand data for failure to start, which is not consistent with the SR. This also creates inconsistency with the uncertainty parameters.	DA-A3	**OPEN**: The scope of this issue is limited to reliability data for standby components. The current treatment approximates standby reliability parameters. The impact of this issue on the surveillance frequency control program is judged to be small. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
DA-C5-01	The update process does not specifically ensure that short term (most all) data is applied to the Bayes update appropriately. Also, the basis documentation incorrectly combines Binomial and Exponential data using a simple mathematical combination.	DA-C5	**OPEN**: The scope of this issue is limited to reliability data for standby components. The current treatment approximates standby reliability parameters. The impact of this issue on the surveillance frequency control program is judged to be small. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with

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			the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
DA-C10-01	Could not find evidence that the guidance was followed when using surveillance procedures.	DA-C10	**OPEN**: Unavailability data collection was performed under procedures that have now been superseded at DESC as part of integration into the Dominion Fleet. There are no known deficiencies in unavailability data collection. The magnitude of the impact of this issue on the surveillance frequency control program is judged to be small. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
DA-D5-01	The exclusion of higher order CCF terms as proposed in Attachment 3 is not a valid approach since the excluded terms represent part of the overall failure rate. DA.3 does provide a sensitivity. This may demonstrate acceptability for an application but does not correct the model.	DA-D5	**OPEN**: The scope of this issue is limited to a small number of CCF groups in the PRA model. A sensitivity analysis has been performed to show that the impact of this issue on total CDF and LERF is small. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
HR-G7-01	An HEP (HEP-C-ALTCOOL-CH) was identified as having been omitted from the dependency analysis and retained at the nominal probability during quantification with no justification. With a Tsw of ~46 minutes, HEP-C-ALTCOOL-CH was not subjected to a dependency analysis, although it appears in cutsets with various combinations (e.g., HEP-DEP-6, -8, -18, -19, -21) and other HFEs (e.g., HEP-C-SGI, HEP-C-TRIPRCP). A PRA tracker item PRACC 19300) self-identified this deficiency along with another HEP (HEP-C-FTS-ESF) omitted for LERF dependency.	HR-G7	**OPEN**: The Scope of this issue is limited to HEP-C-ALTCOOL-CH event in the PRA model. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.

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FOID	**Description (Note: In this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
02-03	Within the SR mapping tables in the various analyses, the N/A entries should point to documentation that discusses why the SR is N/A. (This F&O originated from SR IE-D1)	IE-D1	**OPEN**: This issue does not affect quantification of the VCSNS PRA model. Therefore, there is no impact on the Surveillance Frequency Control Program.
02-05	Documentation that plant personnel (e.g., operations, maintenance, engineering, safety analysis) have been interviewed to determine if potential initiating events have been overlooked could not be located. (This F&O originated from SR IE-A8)	IE-A8	**RESOLVED**: An operator interview has been performed and documented to review plant operating experience related to initiating events. This issue is considered resolved and therefore has no impact on the surveillance frequency control program.
02-06	Not all of the assumptions are documented in the Assumptions section (e.g., pressurizer relief valve sticks open if SI is not secured following a secondary sideline break, many of the ISLOCA assumptions). (This F&O originated from SR IE-D3)	IE-D3	**OPEN**: This issue does not affect quantification of the VCSNS PRA model. Therefore, there is no impact on the Surveillance Frequency Control Program.
02-12	CN-RAM-14-030 does not document the development of the fault tree initiator models or the quantification of the SSIEs. (This F&O originated from SR IE-C9.)	IE-C9	**OPEN**: This issue does not affect quantification of the VCSNS PRA model. Therefore, there is no impact on the Surveillance Frequency Control Program.
02-15	CN-RAM-14-033, Table 7.1-1 contains a list of procedures that were reviewed, organized by system. However, the list does not cover all modeled systems. For example, no procedures for Electric Power, RBCU, HVAC or the Pressurizer Pressure Relief System are included. There should be some discussion since the system notebooks point back to the HRA to justify that there are no pre-initiators for those systems. (This F&O originated from SR HR-A1.)	HR-A1	**PARTIALLY RESOLVED**: The PRA model systems analysis has been reviewed to confirm that the modeling is appropriate. All identified changes to the PRA model have been applied to the model, the outstanding resolution of the issue is to complete the model documentation. Therefore, this issue has no impact on the Surveillance Frequency Control Program.
02-17	Procedures appear to have been screened from further review at too high a level based on known faulty screening criteria as discussed in the Peer Review lessons learned document, using bases such as: "Assume post maintenance testing would reveal any errors," "Includes verified restoration line-up," or "Unlikely to render the system inoperable." NUREG-1792, section 4.2.3.1, which was used as a basis for screening, requires that there be a compelling signal "(e.g., annunciator or indication) of improper equipment status or inoperability in the control room, it is checked at least once per shift or once per day, and realignment can be easily accomplished." Also, in some cases the screening process appears not to have been consistently applied. Actions FBVCC------HE, OACSAVXVG8153FC and OACSAVXVG8154FC	HR-B1	**PARTIALLY RESOLVED**: The PRA model systems analysis has been reviewed to confirm that the modeling is appropriate. All identified changes to the PRA model have been applied to the model, the outstanding resolution of the issue is to complete the model documentation. Therefore, this issue has no impact on the Surveillance Frequency Control Program.

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	were not screened even though they were related to SOP procedure while all other SOP procedures in the table were screened. (This F&O originated from SR HR-B1.)		
03-07	No consideration for outliers considered. No distinction made between valves that are frequently operated and those infrequently operated. SR mapping claims this is N/A. (This F&O originated from SR DA-B2)	DA-B2	**RESOLVED**: Processes and documentation related to the data element of VCS PRA have been updated using Dominion fleet standard approaches. The model documentation now discusses the approach to outliers. No plant-specific data in the VCSNS PRA was disregarded for any reason, there were no outliers identified within data grouping. Therefore, this issue has no impact on the Surveillance Frequency Control Program.
04-05	Utility used mixture of MAAP5 runs and control room simulation runs to support their success criteria, HRA timing and any assumptions they used in PRA. While MAAP5 has been widely used in PRA and is completely appropriate, the use of the operator simulations runs are questionable, especially with no documentation available to review the inputs and outputs, and no maintenance program for maintenance in accordance with the PRA Standard (for example, has it been benchmarked against other software and has it been used within its known limits of applicability). (This F&O originated from SR SC-B4)	SC-B4	**OPEN**: All HRA Time windows have been reanalyzed with MAAP5. System success criteria are supported with a combination of design basis information and MAAP analysis. The remaining scope of this issue is to establish a clear cross-reference between the thermal-hydraulic cases and the success criteria/HRA time windows they support. It is anticipated that completely resolving this issue will have no impact on the Surveillance Frequency Control Program.
04-07	The documentation success criteria do not facilitate the peer review since navigation in the notebooks is a real challenge due to lack of cross referencing. (This F&O originated from SR SC-C1)	SC-C1	**OPEN**: This issue does not affect quantification of the VCSNS PRA model. Therefore, there is no impact on the Surveillance Frequency Control Program.
04-08	Most of the items in this SR are not met. For example, identification of calculation and what they support, a description of limitations, basis for establishing time window for operator actions, etc. (This F&O originated from SR SC-C2)	SC-C2	**OPEN**: This issue does not affect quantification of the VCSNS PRA model. Therefore, there is no impact on the Surveillance Frequency Control Program.
04-24	No evidence could be found on evaluation of the impact of containment seals, penetrations, hatches etc. on containment ultimate capacity. (This F&O originated from SR LE-D2)	LE-D2	**OPEN**: Containment ultimate capacity basis was reviewed to confirm that containment seals, penetrations, hatches were not limiting or compromising containment ultimate capacity. Therefore, this issue has no impact on the Surveillance

Internal Events PRA F&Os and SFCP Dispositions			
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			Frequency Control Program. This finding is considered open until model documentation is finalized.
04-32	Limits of applicability are documented in Section 8.3. However, this documentation appears to be minimum; more detailed documentation is needed to meet the intent of this SR. (This F&O originated from SR LE-G5)	LE-G5	**OPEN**: This item was reviewed during the 2020 Appendix X closeout activity. The independent assessment team concluded that additional detail is required to resolve this issue. This issue does not affect quantification of the VCS PRA model. Therefore, there is no impact on the Surveillance Frequency Control Program.
05-22	Event trees and discussion for the SSB (in containment, Sec. 7.9) and SLOCAs (Sec. 7.10) include RBCUs to succeed, but these are not required for MLOCA (7.8) and LLOCA (Sec 7.5). It is unclear why they are required for prevention of core damage for these event, for MLOCA and LLOCA the document states they are only needed for containment pressure control and integrity so these should also apply to at least the SLOCAs and the SSBs (in containment). (This F&O originated from SR AS-A6)	AS-A6	**PARTIALLY RESOLVED**: The PRA model Success Criteria has been reviewed to confirm that the modeling is appropriate. No changes to the PRA model are required to resolve this issue. Therefore, this issue has no impact on the Surveillance Frequency Control Program.
06-07	No review could be found in Flooding documentation to confirm accident sequence applicability to the flood scenario. (This F&O originated from SR IFQU-A1)	IFQU-A1	**OPEN**: An appropriate initiating event (i.e. accident sequence), such as reactor trip, loss of CC, etc. has been assigned for each flood scenario. Additional review and discussion within the flooding documentation is needed to confirm the accident sequence binning. It is anticipated that resolving this issue will have no impact on the Surveillance Frequency Control Program.
06-12	Susceptibility of PRA equipment (by type only) to flood is documented in Section 7.6, Table 7.6-1. However, failures such as jet impingement or pipe whip, for example, are very flood area and source specific, and therefore, first, a more detailed review has to be performed to identify PRA equipment in each flood area, and then identify susceptibility to failure mechanisms. Also, after flood walkdown, it became clear that steam driven EFW pump is also susceptible to failure due to high humidity in the IB due to HELB since the room that contains the pump has a vent opening big enough to impact conditions inside that room. (This F&O originated from SR IFSN-A6)	IFSN-A6	**OPEN**: The flooding model assumed that all PRA-related components within a flooded area will fail as a result of spray, submergence, or other mechanisms. The only exceptions were made for passive components (Heat exchangers, check valves, etc.) and for components whose design, spatial effects, low pressure source potential, or other reasonable judgment could be made on a case-by-case basis to limit the damage state. The Turbine Driven EFW Pump was

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			not exempted. In order to meet Capability Category II of IFSN-A6 only Submergence and Spray failure modes are required. Resolution of this F&O is anticipated to have no impact on the surveillance frequency control program.
06-17	No system design walkdowns were found to support recent modeling updates. (This F&O originated from SR SY-A4)	SY-A4	**OPEN**: Walkdowns have been conducted for Fire, Seismic and Flooding PRA development between 2014 and present. No additional insights from documenting internal events-specific walkdowns are anticipated. Resolution of this F&O is anticipated to have no impact on the surveillance frequency control program.
06-22	The internal flooding model indicates that isolated flooding events will be accounted for in the internal events modeling. However, no contribution was defined in the initiating event frequency of the internal events. Therefore, the isolation of CCW or SW given a successful flood isolation (state FS02 in the IF notebook) is not present in the model to cause an isolation of the system response model. The documentation does not sufficiently support successful operation of the CSIP or RHR pumps given a lack of room cooling. The analysis from the IPE includes door opening to demonstrate marginal performance for the CSIP room when no HVAC is present. No HVAC is modeled or discussed in the CSIP notebook and no compensatory HFE was found in the HRA notebook. (This F&O originated from SR SY-A18)	SY-A18	**OPEN**: The VCSNS PRA does not credit isolation of pressure boundary failures in the CC and SW systems, no changes to PRA model/quantification are required to resolve this issue. Flooding documentation update is required to clarify flood isolation actions and effect on mitigating equipment. Therefore, resolution of this issue has no impact on the Surveillance Frequency Control Program. Refer to similar finding 06-23 for the disposition of CSIP and RHR pumps room cooling modeling.
06-23	The supporting HVAC information from notebook documentation (CN-COA-91-129) indicates that "Among other things, the HVAC system provides cooling of the RHR/Spray Pump room and the Charging/SI pump room. The other rooms inside the intermediate building have been eliminated from consideration. An analysis was performed to determine if a failure of HVAC would cause a subsequent failure of the pumps located within these 2 rooms. The analysis results, presented in reference 8, indicate that HVAC is not required as long as the doors are open to provide natural circulation cooling in these pump rooms."	SY-A22	**PARTIALLY RESOLVED**: The justification for HVAC success criteria for charging/SI pump rooms and RHR/Spray Pump rooms was reviewed and it was determined that no changes to the PRA model are required. Documentation update is required to clarify HVAC dependencies on mitigating equipment. Resolution of this issue does not impact the PRA model; therefore, this issue does not impact the Surveillance Frequency Control Program.

Internal Events PRA F&Os and SFCP Dispositions			
Part 3 – F&Os with No Impact to SFCP			
FOID	**Description (Note: In this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
	This appears to indicate that some form of cooling is required for the CSIP and RHR/Spray pumps. No HVAC cooling requirement is documented in the ECCS notebook and not operator actions are defined to open doors for these rooms. (This F&O originated from SR SY-A22)		
06-25	HVAC has been screened for pump rooms in some cases based on the expected conditions in the room with the door open. Given that the doors are normally closed, and that no operator action is modeled the implication is that they are essentially taking credit for an unmolded action. (This F&O originated from SR SY-B12)	SY-B12	**PARTIALLY RESOLVED**: The basis for HVAC success criteria was reviewed to confirm that, if required, operator actions are modeled to establish compensatory actions for loss of room cooling. Documentation update is required to clarify HVAC dependencies and success criteria in the PRA model. Resolution of this issue does not impact the PRA quantification; therefore, this issue does not impact the Surveillance Frequency Control Program.
06-27	There are several canned statements in all of the notebooks. For example: 1) The systems and components credited in the PRA model are expected to function in the environments anticipated following events and these ensuing environments are not expected to exceed component qualifications. 2) The environmental conditions are expected to be similar to those expected for design basis events. 3) No environmental conditions are expected that will exceed the operating qualifications of the components. 4) No non-qualified equipment is credited for operation in harsh environments. 5) The impact of high energy line breaks is addressed in the internal flooding analysis. Systems and components are credited for conditions and loads expected during the initiating events modeled. (This F&O originated from SR SY-B14)	SY-B14	**OPEN**: This finding relates to documentation of HVAC dependencies. Clarification of the relationship between EQ requirements for front line systems and PRA modeled HVAC is required. Resolution of this issue has no impact on the Surveillance Frequency Control Program.
06-28	The documentation performed a qualitative screening and did not estimate any contribution due to HIF. This is inconsistent with the EPRI guidance documented used to develop the internal flooding assessment. (This F&O originated from SR IFEV-A7)	IFEV-A7	**OPEN**: In addition to the limited maintenance activities, EPRI guidelines for internal flooding probabilistic risk assessment (EPRI 1019194 section 5.6) allow qualitative screening if two or more isolation valves are used. Station administrative procedure SAP-0201 states that two valve

Internal Events PRA F&Os and SFCP Dispositions			
Part 3 – F&Os with No Impact to SFCP			
FOID	**Description (Note: In this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**SRs**	**SFCP Disposition**
			protection is required for systems or tanks that can cause major flooding of buildings or systems being worked. Based on this, qualitative screening of (HIF) human induced flooding is considered to be generally appropriate in the VCSNS internal flooding analysis. Formal review of human induced flooding events as proscribed by EPRI-1019194 is required in order to resolve this issue. The resolution of this issue is expected to have no impact on the Surveillance Frequency Control Program.

4.2 Fire PRA F&Os

Fire PRA F&Os and SFCP Dispositions			
FOID	**Description (Note: in this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**Supporting Requirement**	**SFCP Disposition**
CF-A1-01	(Note: This F&O is based on the original peer review.) Attachment 8 to DC00340-001, Circuit Failure Mode Likelihood Analysis, Task 5.10, documents the results of the circuit failure analyses and assigns failure probabilities to specific cable failure modes. NUREG/CR-6850 Section 10.5.2 provides 2 recommended options for assigning CF probability values. Option 1 (use of tables) is recommended when circuits are of a type bounded by circuit testing, which includes grounded circuit. Option 2, The probability estimate formulas, are recommended for cases "where: * The circuit is ungrounded or is impedance grounded without ground fault trip capability." Contrary to the recommendations, the use of tables was used for all circuit types in Attachment 8 to DC00340-001, without a justification for the use of this process. In addition, cable failure likelihood values assigned in Attachment 8 to DC00340-001 do not always reflect Section 2.0 "Scope/Methodology" (which is based on NUREG/CR-6850, Vol. 2, Chapter 10) and the rationale for using different values is not documented in the calculation. Specifically, Section 1 of Attachment 8 to DC00340-001 and Section 10.5.2 of NUREG/CR-6850, include criteria for the appropriate use of the Tables 10-1 - 10-5 of NUREG/CR-6850: The circuit is of a grounded design. NUREG/CR-6850 Vol. 2 Section 10.5.2 states that: "The probability estimate formulas are recommended for cases where: ....* The circuit is ungrounded or is impedance grounded without ground fault trip capability, " Components addressed in Attachment 8 to DC00340-001 include ungrounded dc circuits, contrary to the statements in Section 2 of the calculations. No justification is provided for using the tabular values (as opposed to the Computational Probability Estimates of NUREG/CR-6850 for ungrounded circuits. In addition, It appears that a 0.30 was used as a default value for Psacd in Attachment 8 to DC00340-001 Rev. A as a highest screening value. This value is based on the presence of a CPT in Task 10 of NUREG/CR-6850 (which would apply to MOVs. Tables 10-2 and 10-4 of NUREG/CR-6850 Vol. 2 show a best estimate of 0.60 for M/C intra-cable thermoplastic cables without CPT. Use of the values that are inconsistent with industry guidance without justification will result in inconsistent results and future issues with program configuration control.	CF-A1	**PARTIALLY RESOLVED**: The treatment of Circuit Failure Mode Likelihood Analysis (CFMLA) was updated following peer review, addressed using industry methods, explained to the NRC via RAI process, and accepted in the 2015 SE. During the 2018 timeframe, updates were made to risk significant circuits to use newer treatment in accordance with NUREG/CR-7150 Vol. 2. Outstanding effort to completely resolve this issue includes dispositioning instances where PRA documentation and the PRA model are not in complete alignment and documenting extent and basis for partial implementation of NUREG/CR-7150. The impact of this issue will be confirmed on a surveillance evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolve.

Fire PRA F&Os and SFCP Dispositions			
FOID	**Description (Note: In this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**Supporting Requirement**	**SFCP Disposition**
	Address circuit failure probabilities using the recommended process in NUREG/CR-6850 or provide a technical basis for use of plant-specific values.		
ES-B1-01	(Note: This F&O is based on the original peer review.) The development of the Fire PRA is very data intensive and much of the work associated with the quantification process is entirely dependent of the validity of data linkages in the various databases. The key analysis databases are PC-CKS and FRANX. A review of the Fire PRA found numerous data inconsistencies and linkage issues between these two files. In addition, it appears that other key data relationships that are critical to the analysis do not exist in these two databases - suggesting that there are other key sources of data that are needed. The review of the key databases found instances where data from PC-CKS and FRANX are not properly coordinated. These are generally reflected in the various tables ultimately referring to PRA model basic events that do not exist. As a consequence, while the developed data (equipment and cable listings) indicate that certain fire induced failures are treated in the Fire PRA, the data inconsistencies would result in these elements not being propagated into the actual quantification of the PRA model. Another very key concern is the treatment of fire induced spurious replacements in FRANX. Based on discussions and a review of FRANX, it appears that this data is entirely developed manually - not via a database query. In addition, the resulting table and associated documentation does not retain the data linkages to PC-CKS. Several errors were identified in the development of this table in FRANX - again causes errors in the propagation of fire induced effects. It is suggested that a comprehensive confirmation of data integrity and consistency be performed and that any required intermediate translation tables, data relationships, or queries be identified and integrated into the project documentation and analysis files.	ES-B1	**PARTIALLY RESOLVED**: The related documentation was reviewed during 2020 Finding closeout activity. The Independent assessment team concluded that "Questions about the fidelity in mapping fire-induced component failures to Basic Events in the Fire PRA (as described in Findings ES-B1-01 and ES-B1-03) continue to persist because of certain anomalies between the documentation and the databases." These anomalies are judged to have very small impact on the surveillance frequency control program. The impact of this issue will be confirmed on a surveillance evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
ES-B1-03	(Note: This F&O is based on the original peer review.) The treatment (crediting) of components in the Fire PRA depends largely on the manner in which individual PRA model basic events are linked to spatial data via FRANX and PC-CKS. A review of the data found that out of about 2,800 PRA model basic events, less than 900 are mapped to spatial data and used to control the quantification process. The remaining unmapped PRA model basic events include many items that represent component failure modes that could be	ES-B1	**PARTIALLY RESOLVED**: The related documentation was reviewed during 2020 Finding closeout activity. The Independent assessment team concluded that "Questions about the fidelity in mapping fire-induced component failures to Basic Events in the Fire

Fire PRA F&Os and SFCP Dispositions			
FOID	**Description (Note: In this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**Supporting Requirement**	**SFCP Disposition**
	induced by a fire. While it is possible that all of these have been effectively subsumed by the mapped basic events, in the absence of some documentation or explicit treatment, it is not possible to ascertain that these unmapped events have not inadvertently been credited in the quantification. The potential that random basic events could be included in the Fire PRA quantification when they should have otherwise been set to TRUE could result in invalid results (low CCDP). An effort should be undertaken and documented to demonstrate that the Fire PRA only relies on those functional features of the VC Summer plant for which spatial equipment and cable location data is developed.		PRA (as described in Findings ES-B1-01 and ES-B1-03) continue to persist because of certain anomalies between the documentation and the databases." These anomalies are judged to have very small impact on the surveillance frequency control program. The impact of this issue will be confirmed on a surveillance evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
HRA-B4-01	(Note: This F&O is based on the original peer review). The Evaluation of EOPs for Undesired Operator Actions per Table 6c of DC00340-001 depicts Instrumentation TI-499A and TI-499B as not screened since the EOP's say to check TI-499A and TI-499B only, for RCS Sub cooling. Instrumentation TI-499C/D are specifically excluded per the documentation; however, these two instruments are included under the "AND" gate G320. This issue relates to an issue of the documentation not matching the model and an error in the modeling. Correct the Fault Tree logic and ensure that documentation matches the logic.	HRA-B4	<u>**OPEN**</u>: This issue is judged to have very small impact on the surveillance frequency control program because of the limited scope of this issue. The impact of this issue will be confirmed on a surveillance evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
HRA-B4-02	(Note: This F&O is based on the original peer review.) Logic under gate G317 includes three different types of instrument failures: temperature transmitters, level transmitters, and pressure transmitters. The level and temperature transmitters are discussed in the documentation Attachment 2 to DC00340-001 task 5.2, Table 6.2C, however, the pressure transmitters are not discussed. This is a gap between the documentation and the fault tree database. In addition, neither the pressure transmitter nor the level transmitter is listed in Table 6d-3. Ensure that the model and the documentation match.	HRA-B4	<u>**OPEN**</u>: This issue is judged to have very small impact on the surveillance frequency control program because of the limited scope of this issue. The impact of this issue will be confirmed on a surveillance evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
HRA-C1-01	(Note: This F&O is based on the original peer review.) The timing evaluation for Operator Action, BAPM-XPP39AHE-F (Operator Fails to start SW pump P-39A) is based upon an operator action to swap charging pumps in order to gain additional time for this HRA. In essence, an HRA within an HRA exists with no accounting for the failure dependencies associated with swapping	HRA-C1	<u>**OPEN**</u>: This issue is judged to have very small impact on the surveillance frequency control program because of the limited scope of this issue. The impact of this issue will be confirmed on a

Fire PRA F&Os and SFCP Dispositions			
FOID	**Description (Note: In this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**Supporting Requirement**	**SFCP Disposition**
	the charging pumps. The dependencies associated with the operator action to swap charging pumps is relatively large and is not accounted for this analysis. Remove the dependency for the charging pump swap in the recovery action.		surveillance evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
PRM-B9-02	(Note: This F&O is based on the original peer review.) Upon examination, selected components identified in ES (Task 5.2, table 3a) for inclusion into the PRA could not be validated as having been incorporated into the model. (example: FCV-0122). The linkage between ES and PRM is critical to assuring appropriate quantification results. Review the items in Table 3a and provide a clear disposition and link to the treatment of these items in PRM.	PRM-B9	**PARTIALLY RESOLVED**: The related documentation was reviewed during 2020 Finding closeout activity. The Independent assessment team concluded that "This F&O is redundant and is dependent on the resolution of ES-B1-03". Therefore, refer to finding ES-B1-03 for disposition of the impact of this finding on the Surveillance Frequency Control Program.

4.3 Seismic PRA F&Os

Seismic PRA F&Os and SFCP Dispositions			
FOID	**Description (Note: in this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**Supporting Requirement**	**SFCP Disposition**
19-10	The previous IE-PRA peer review generated more than 90 Finding-level F&Os that had not been resolved at the time of the SPRA peer review. These F&Os are documented in the Self Assessment Report for SPR (LTR-RAM-18-15), along with possible resolutions and the potential impact of the possible resolutions on the SPRA model. Because of the broad nature of the F&Os against the IE-PRA model and the lack of actual resolutions, it was not possible for the SPRA Peer Review Team to assess the collective impact of the open F&Os on the SPRA model. Several examples are offered: Findings written against SRs HR-F2, G4, G5, & G6 address timing input to HRA. Timing is one of the key inputs to adjusting HEPs for seismic impact. Findings written against SRs SC-B3, B4, & B5 address the basic success criteria in the IE-PRA. This can directly impact accident sequences in the SPRA model. Findings written against SY-A22, B6, & B12 address issues of modeling HVAC systems. The absence of these systems in the IE-PRA carries over directly to the SPRA. (This F&O originated from SR SPR-B1)	SPR-B1	<u>**OPEN**</u>: This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
19-07	FLEX equipment is not modeled in the SPRA. Not taking credit for FLEX may result in an overly conservative model. In particular, FLEX equipment may be important to realistically address the safe-stable state for a seismic event (see related F&O 26-2). (This F&O originated from SR SPR-C1)	SPR-C1	<u>**OPEN**</u>: Conservatively excluding FLEX from the SPRA model applies conservative bias to risk calculations and is therefore acceptable for use in the Surveillance Frequency Control Application, which utilizes absolute risk metrics (i.e. delta CDF, delta LERF) as acceptance criteria rather than metrics which are relative to total CDF/LERF such as RAW, Fussell-Vesely, etc. If it is determined that this approach causes a surveillance frequency evaluation to fail to meet its PRA acceptance criteria or is otherwise unable to generate meaningful results, then the PRA model will be reviewed and revised to include the level of detail needed to complete the evaluation in accordance with NEI 04-10 Steps 9 & 11.

Seismic PRA F&Os and SFCP Dispositions			
FOID	**Description (Note: in this column, "SR" denotes "Supporting Requirement", used in the original F&O description.)**	**Supporting Requirement**	**SFCP Disposition**
24-07	The liquefaction potential was not considered in identification of failure modes that can affect the Service Water system. (This F&O originated from SR SFR-D1)	SFR-D1	**OPEN**: The impact of resolving this issue on the surveillance frequency control program is judged to be small. This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.
20-01	The PSHA for the VC Summer site was performed using the existing seismic source model described in NUREG-2115. HLR SHA-H, as modified in Regulatory Guide 1.200, states 'ENSURE, in light of established current information, the study meets the requirements in HLR-SHA-A thru HLR-SHA-G.' The NUREG-2115 source model was completed in 2012, using an earthquake catalog for the time period ending in 2008. (This F&O originated from SR SHA-H1)	SHA-H1	**OPEN**: This issue will be reviewed and assessed on an evaluation-specific basis in accordance with the NEI 04-10 process (Steps 5, 14) until this issue is considered resolved.

ATTACHMENT 3

EXISTING TS PAGES MARK-UP

Virgil C. Summer Nuclear Station (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

INSERT 1

in accordance with the Surveillance Frequency Control Program

INSERT 2

o. Surveillance Frequency Control Program

This program provides control 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 Operations are met.

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

2). 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.

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

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.
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.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.77% delta k/k for 3 loop operation.

APPLICABILITY: MODES 1, and 2*.

ACTION:

With the SHUTDOWN MARGIN less than 1.77% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.77% delta k/k:

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

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at ~~least once per 12 hours~~ by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits 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 Surveillance Requirement 4.1.1.1.2 with the control banks at the maximum insertion limit of Specification 3.1.3.6.

*See Special Test Exception 3.10.1

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

INSERT 1

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k ~~at least once per 31 Effective Full Power Days (EFPD)~~. This comparison shall consider at least the following factors:

1. Reactor Coolant System boron concentration,
2. Control rod position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy production,
5. Xenon concentration, and
6. Samarium.

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 - MODES 3, 4 AND 5

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limits shown in Figure 3.1-3.

APPLICABILITY: MODES 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than the required value, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or 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 the required value:

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

INSERT 1

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

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

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid tanks via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

INSERT 1

SURVEILLANCE REQUIREMENTS

4.1.2.1.1 At least one of the above required flow paths shall be demonstrated OPERABLE ~~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.

4.1.2.1.2 Demonstrate operability of the required charging pump per Surveillance 4.5.2.f.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. The flow path from the boric acid tanks via a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

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

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2 percent delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

INSERT 1

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

b. ~~At least once per 18 months~~ by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 30 gpm to the Reactor Coolant System.

#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum contained borated water volume of 2700 gallons,
2. Between 7000 and 7700 ppm of boron, and
3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 51,500 gallons,
2. A minimum boron concentration of 2300 ppm, and
3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

INSERT 1

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

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. ~~At least once per 24 hours~~ by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 40°F.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

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

1. Verifying the boron concentration in the water,

2. Verifying the contained borated water volume of the water source, and

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

b. ~~At least once per 24 hours~~ by verifying the RWST temperature when the outside air temperature is less than 40°F.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

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

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

INSERT 1

SURVEILLANCE REQUIREMENTS

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

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

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The shutdown and control rod position indication system and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

   1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

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

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

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

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

SURVEILLANCE REQUIREMENTS

INSERT 1

4.1.3.2 Each rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 12 steps ~~at least once per 12 hours~~ except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system at least once per 4 hours.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

INSERT 1

4.1.3.3 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of an ANALOG CHANNEL OPERATIONAL TEST ~~at least once per 18 months~~.

*With the reactor trip system breakers in the closed position.
#See Special Test Exception 3.10.5.

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

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

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

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

INSERT 1

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

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Restore the rod to within the limit specified in the COLR, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR.

INSERT 1

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

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

*See Special Test Exceptions 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.0

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR) figure entitled Rod Group Insertion Limits versus Thermal Power For Three Loop Operation.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

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

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

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

SURVEILLANCE REQUIREMENTS

INSERT 1

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

*See Special Test Exceptions 3.10.2 and 3.10.3
#With K_{eff} greater than or equal to 1.0.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

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

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

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

INSERT 1

4.2.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 When in Base Load operation, the target flux difference of each OPERABLE excore channel shall be determined by measurement ~~at least once per 92 Effective Full Power Days~~. The provisions of Specification 4.0.4 are not applicable.

INSERT 1

4.2.1.4 When in Base Load operation, the target flux difference shall be updated ~~at least once per 31 Effective Full Power Days~~ by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map

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

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

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

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

c. Satisfying the following relationship:

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

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

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing tolerances and measurement uncertainty as specified in the COLR, F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z) and W(z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

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

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined, * or

INSERT 1

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

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

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND} when the Power Distribution Monitoring System (PDMS) is inoperable; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS at any THERMAL POWER greater than APL^{ND}; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

c. Satisfying the following relationship:

$$F_Q^M(z) \le \frac{F_Q^{RTP} \times K(z)}{P \times W(z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing and measurement uncertainties as specified in the COLR. The F_Q limit is F_Q^{RTP}. P is the relative THERMAL POWER. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. F_Q^{RTP}, K(z) and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

d. Measuring $F_Q^M(z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering BASE LOAD operation after satisfying Section 4.2.2.3 unless a core power distribution measurement has been obtained in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to measurement, and

INSERT 1

2. ~~At least once per 31 Effective Full Power Days.~~

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through a core power distribution measurement and RCS total flow rate comparison, to be within the region of acceptable operation specified in the COLR prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation specified in the COLR.

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

INSERT 1

~~b. At least once per 31 Effective Full Power Days.~~

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation specified in the COLR ~~at least once per 12 hours~~ when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

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

4.2.3.5 The RCS total flow rate shall be determined by heat balance measurement at ≥90% RATED THERMAL POWER ~~at least once per 18 months.~~

ACTION: (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS INSERT 1

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

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

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 percent RATED THERMAL POWER with one Power Range Channel inoperable ~~at least once per 12 hours~~ by using the PDMS or movable incore detectors to confirm that the normalized symmetric power distribution is consistent with the indicated QUADRANT POWER TILT RATIO. The incore detector monitoring shall be done with 2 sets of 4 symmetric thimbles or a full incore flux map.

POWER DISTRIBUTION LIMITS

3/4 2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

INSERT 1

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor trip system instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by performance of the reactor trip system instrumentation surveillance requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be verified to be within ~~its limit at least once per 18 months. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.~~

Insert 1

Replace each marked through surveillance frequency in the Check, Calibrate, and Test columns with "SFCP"

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A	N.A.	~~R~~(11)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux High Setpoint	~~S~~	~~D~~(2, 4), ~~M~~(3, 4), ~~Q~~(4, 6), ~~R~~(4, 5)	~~SA~~	N.A.	N.A.	1, 2
Low Setpoint	~~S~~	~~R~~(4)	S/U(18), (16)	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux High Positive Rate	N.A.	~~R~~(4)	~~SA~~	N.A.	N.A.	1, 2
4. Deleted						
5. Intermediate Range, Neutron Flux	~~S~~	~~R~~(4)	S/U(18), (16)	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	~~S~~	~~R~~(4)	S/U(18), (17), (9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	1, 2
8. Overpower ΔT	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	1
10. Pressurizer Pressure--High	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	1
12. Loss of Flow	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level–Low-Low	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	1,2
14. Steam Generator Water Level - Low Coincident with Steam/Feedwater Flow Mismatch	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.	~~R~~	N.A.	~~SA~~	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	~~R~~	N.A.	~~SA~~	N.A.	1
17. Turbine Trip						
A. Low Fluid Oil Pressure	N.A.	~~R~~	N.A.	(1, 8, 10)	N.A.	1
B. Turbine Stop Valve Closure	N.A.	~~R~~	N.A.	(1, 8, 10)	N.A.	1
19. Reactor Trip System Interlocks						
A. Intermediate Range Neutron Flux, P-6	N.A.	~~R~~(4)	~~R~~	N.A.	N.A.	2##
B. Low Power Reactor Trips Block, P-7	N.A.	~~R~~(4)	~~R~~	N.A.	N.A.	1
C. Power Range Neutron Flux, P-8	N.A.	~~R~~(4)	~~R~~	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT		CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
D	Low Setpoint Power Range Neutron Flux, P-10	N.A.	~~R~~(4)	~~R~~	N.A.	N.A.	1,2
E.	Turbine Impulse Chamber Pressure, P-13	N.A.	~~R~~	~~R~~	N.A.	N.A.	1
F.	Low Power Range Neutron Flux, P-9	N.A.	~~R~~(4)	~~R~~	N.A.	N.A.	1
20.	Reactor Trip Breaker	N.A.	N.A.	N.A.	(7, 12)	N.A.	1, 2, 3,* 4*, 5*
21.	Automatic Trip Logic	N.A.	N.A.	N.A.	N.A.	~~Q~~ (15)	1, 2, 3*, 4*, 5*
22.	Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	(7, 13), ~~R~~(14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

*	-	With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.
##	-	Below P-6 (Intermediate Range Neutron Flux Interlock) setpoint.
###	-	Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint.
(1)	-	If not performed in previous 31 days.
(2)	-	Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(3)	-	Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(4)	-	Neutron detectors may be excluded from CHANNEL CALIBRATION.
(5)	-	Detector plateau curves shall be obtained evaluated and compared to manufacturer's data. For the Power Range Neutron Flux Channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(6)	-	Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(7)	-	Each train shall be tested ~~at least every 124 days on a STAGGERED TEST BASIS.~~ Insert 1
(8)	-	Prior to entering MODE 1 whenever the unit has been in MODE 3.
(9)	-	Surveillance in MODES 3*, 4* and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.
(10)	-	Setpoint verification is not required.
(11)	-	The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).
(12)	-	The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.
(13)	-	Local manual shunt trip prior to placing breaker in service.
(14)	-	Automatic undervoltage trip.
(15)	-	Each train shall be tested ~~at least every 184 days on a Staggered Test Basis.~~ Insert 1
(16)		12 hours after reducing power below P-10 and ~~184 days thereafter~~.
(17)		4 hours after reducing power below P-6 ~~and 4 hours after entering MODE 3 from MODE 2 and 184 days thereafter~~.
(18)		If not performed in previous 184 days.

INSTRUMENTATION

3/4 3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint Column but more conservative than the value shown in the Allowable Value Column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value Column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to its OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

c. With an ESFAS instrumentation channel or interlock inoperable take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the engineered safety feature actuation system instrumentation surveillance requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS ~~function shall be verified to be within the limit at least once per 18 months. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.~~

Insert 1

Replace each marked through surveillance frequency in the check, calibration, and test column with "SFCP".

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER								
a. Manual Initiation	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~Q~~(1)	~~Q~~(1)	~~R~~(3)	1, 2, 3, 4
c. Reactor Building Pressure-High-1	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure–Low	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines–High	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Line Pressure Low	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. REACTOR BUILDING SPRAY								
a. Manual Initiation	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~Q~~(1)	~~Q~~(1)	~~R~~(3)	1, 2, 3, 4
c. Reactor Building Pressure-High-3	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continue)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. STEAM LINE ISOLATION								
a. Manual	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~Q~~(1)	~~Q~~(1)	~~R~~(3)	1, 2, 3
c. Reactor Building Pressure-High-2	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure Low	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION								
a. Steam Generator Water Level--High-High	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2
b. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	~~Q~~(1)	~~Q~~(1)	~~R~~(3)	1, 2
6. EMERGENCY FEEDWATER								
a. Manual	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~Q~~(1)	~~Q~~(1)	~~R~~(3)	1, 2, 3
c. Steam Generator Water Level--Low-Low	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
EMERGENCY FEEDWATER (Continued)								
d. Undervoltage - Both ESF Busses	N.A.	~~R~~	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						
f. Undervoltage - One ESF Bus	N.A.	~~R~~	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2
h. Suction transfer on low pressure	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. LOSS OF POWER								
a. 7.2 kV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	~~R~~	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 7.2 kV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	~~R~~	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3, 4
8. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP								
a. RWST level low-low	~~S~~	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~Q~~(1)	~~Q~~(1)	~~R~~(3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERA-TIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS								
a. Pressurizer Pressure, P-11	N.A.	~~R~~	~~SA~~	N.A.	N.A	N.A.	N.A.	1, 2, 3
b. Low, Low Tavg, P-12	N.A.	~~R~~	~~SA~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION Insert 1

(1) Each train shall be tested ~~at least every 184 days on a STAGGERED TEST BASIS.~~

(2) The 36 inch containment purge supply and exhaust isolation valves are sealed closed during Modes 1 through 4, as required by TS 3.6.1.7. With these valves sealed closed, their ability to open is defeated; therefore, they are excluded from the quarterly slave relay test.

(3) Slave Relay Testing will be conducted Insert 1 ~~every 18 months~~ for Westinghouse type AR relays and preferably during a refueling outage to preclude the risk of actuation. Replacement relays other than Westinghouse type AR or reconciled Cutler-Hammer relays will require further analysis and NRC approval to maintain the established frequency.

Replace each marked through frequency with "SFCP".

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	~~S~~	~~R~~	~~M~~	*
b. Deleted				
2. PROCESS MONITORS				
a. Deleted				
b. Containment				
i. Deleted				
ii. Particulate and Gaseous Activity - RCS Leakage Detection (RM-A2)	~~S~~	~~R~~	~~M~~	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	~~S~~	~~R~~	~~M~~	ALL MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	~~S~~	~~R~~	~~M~~	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	~~S~~	~~R~~	~~M~~	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	~~S~~	~~R~~	~~M~~	1, 2, 3 & 4

* With fuel in the storage pool or building

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO using a full-core flux map per Specification 4.2.4.2, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(z)$.

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE ~~at least once per 24 hours~~, by normalizing each detector output when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(z)$.

Replace each marked through surveillance frequency in the check and calibration column with "SFCP".

INSTRUMENTATION

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Wind Speed Lower 10m	~~D~~	~~SA~~
b. Wind Speed Upper 61m	~~D~~	~~SA~~
2. Wind Direction		
a. Wind Direction Lower 10m	~~D~~	~~SA~~
b. Wind Direction Upper 61m	~~D~~	~~SA~~
3. Atmospheric Stability		
a. Delta T 1 10-61m	~~D~~	~~SA~~
b. Delta T 2 10-40m	~~D~~	~~SA~~

Elevations nominal above grade elevation

Replace each marked through surveillance frequency in the check and calibration column with "SFCP".

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	~~M~~	N.A.
2. Pressurizer Pressure	~~M~~	~~R~~
3. Pressurizer Level	~~M~~	~~R~~
4. Steam Generator Pressure	~~M~~	~~R~~
5. Steam Generator Level	~~M~~	~~R~~
6. Condensate Storage Tank Level	~~M~~	~~R~~
7. Reactor Coolant System Hot Leg Temperature	~~M~~	~~R~~
8. Reactor Coolant System Cold Leg Temperature	~~M~~	~~R~~
9. Reactor Coolant System Pressure	~~M~~	~~R~~
10. Pressurizer Relief Tank Level	~~M~~	~~R~~
11. Reactor Building Temperature	~~M~~	~~R~~
12. Boric Acid Tank Level	~~M~~	~~R~~

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown on Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 30 days or submit a Special Report within the following 14 days from the time the action is required. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to operable status.

b.1 With the number of OPERABLE Reactor Building radiation monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

i) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

ii) Submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

b.2 Deleted

b.3 With the number of OPERABLE accident monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

Insert 1

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performing a ~~monthly~~ CHANNEL CHECK and a CHANNEL CALIBRATION ~~every refueling outage~~. The Reactor Building Radiation Level Instrumentation CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for the range decades above 10R/hr and a single point calibration of the detector below 10R/hr with an installed or portable gamma source.

Replace each surveillance frequency in the Check, Calibration, and Test Columns with "SFCP".

TABLE 4.3-9

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM				
a. Hydrogen Monitor	~~D~~	~~Q~~(1)	~~M~~	**
b. Oxygen Monitor	~~D~~	~~Q~~(2)	~~M~~	**

INSTRUMENTATION

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACTION:

a. With one or more loose part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of:

Insert 1

a. A CHANNEL CHECK ~~at least once per 24 hours~~,

b. An ANALOG CHANNEL OPERATIONAL TEST ~~at least once per 31 days~~, and

c. A CHANNEL CALIBRATION ~~at least once per 18 months~~.

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION (Continued)

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

ACTION:

With any of the operability criteria listed in 3.3.3.11.a, 3.3.3.11.b, or 3.3.3.11.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system, satisfying the OPERABILITY requirements listed in Specification 3.3.3.2, to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11.1 The operability criteria listed in 3.3.3.11.a, 3.3.3.11.b, and 3.3.3.11.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

4.3.3.11.2 Calibration of the PDMS is required:

INSERT 1

a. ~~at least once every 180 Effective Full Power Days~~ when the minimum number and core coverage criteria as defined in 3.3.3.11.b.1 and 3.3.3.11.b.2 are satisfied, or

b. ~~at least once every 31 Effective Full Power Days~~ when only the minimum number criterion as defined in 3.3.3.11.b.3 is satisfied.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All Reactor Coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required Reactor Coolant loops in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENT

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

INSERT 1

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the Reactor Coolant loops listed below shall be OPERABLE and at least one of these Reactor Coolant loops shall be in operation.*

a. Reactor Coolant Loop A and its associated steam generator and Reactor Coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and Reactor Coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and Reactor Coolant pump,

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 coolant 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 OPERABLE ~~once per 7 days~~ by verifying correct breaker alignments and indicated power availability.

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

INSERT 1

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% of wide range indication ~~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

Insert 1

SURVEILLANCE REQUIREMENTS

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

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% of wide range indication ~~at least once per 12 hours~~.

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

4.4.1.3.4 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water ~~at least once per 31 days~~.*

* Not required to be performed until 12 hours after entering MODE 4.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN – LOOPS FILLED

LIMITING CONDITION FOR OPERATION

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

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

b. The secondary side water level of at least two steam generators shall be greater than 10 percent of wide range indication.

APPLICABILITY: MODE 5 with Reactor Coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE and/or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no residual heat removal 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 residual heat removal loop to operation.

SURVEILLANCE REQUIREMENTS

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

Insert 1

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

4.4.1.4.1.3 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water ~~at least once per 31 days~~.

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

A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless 1) the pressurizer water volume is less than 1286 cubic feet and/or 2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

* The RHR pump 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

COLD SHUTDOWN – LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with Reactor Coolant loops not filled.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.4.1.4.2.2 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water ~~at least once per 31 days~~.

Insert 1

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

* The RHR pump may be 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

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1288 cubic feet, (92% of indicated span) and at least two groups of pressurizer heaters each having a capacity of at least 125 kw.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

INSERT 1

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current ~~at least once per 92 days.~~ 14

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

g. With three block valves inoperable:
   1) within 1 hour:
      a) restore the block valves to OPERABLE status, or
      b) place the associated PORVs in manual control and
   2) within the next 2 hours restore at least one of the three block valves to OPERABLE status and
   3) within the next 72 hours:
      a) restore at least two of the three block valves to OPERABLE status and
      b) ensure that the remaining inoperable block valve is closed and the power is removed;

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

h. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE ~~at least once per 18 months~~ by operating the valve through one complete cycle of full travel during MODES 3 or 4. INSERT 1

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

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

detection monitor, analyze grab samples of the containment atmosphere at least once per 12 hours and restore the required reactor building sump level monitor or the reactor building cooling unit condensate flow rate monitor to OPERABLE status within 7 days; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With the required reactor building atmosphere radioactivity monitor and the reactor building cooling unit condensate flow rate monitor inoperable, restore the required reactor building atmosphere radioactivity monitor or the reactor building air cooler condensate flow rate monitor to OPERABLE status within 30 days; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With all required monitoring systems inoperable, enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Reactor building atmosphere particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

INSERT 1

b. Reactor building sump level-performance of CHANNEL CALIBRATION ~~at least once per 18 months~~,

c. ~~Reactor building atmosphere gaseous radioactivity monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION, AND ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified is Table 4.3-3,~~

d. Reactor building cooling unit condensate flow detector-performance of CHANNEL CALIBRATION ~~at least once per 18 months~~.

---

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

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator,

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

e. 33 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. The leakage rate specified for each Reactor Coolant System Pressure Isolation Valve in Table 3.4-1 at a Reactor Coolant System pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any operational Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, primary-to-secondary leakage, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve Leakage greater than the limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the reactor building atmosphere (gaseous or particulate) radioactivity monitor ~~at least once per 12 hours~~.

INSERT 1

REACTOR COOLANT SYSTEM

Insert 1

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the reactor building sump inventory ~~at least once per 12 hours~~.

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

d. Performance of a Reactor Coolant System water inventory balance ~~at least once per 72 hours~~.(1) This requirement is not applicable to primary-to-secondary leakage.

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

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit.

Insert 1

a. During startup ~~following each refueling outage, which may be extended to a performance-based frequency not to exceed 3 refueling outages (to a maximum of 60 months) following 2 consecutive satisfactory tests~~.

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

c. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve for valves denoted on Table 3.4-1 by an asterisk*.

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

4.4.6.2.3 Primary-to-secondary leakage shall be verified ≤ 150 gallons per day through any one steam generator ~~at least once per 72 hours~~.(1)

Insert 1

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

REACTOR COOLANT SYSTEM

TABLE 4.4-3

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	~~At least once per 72 hours~~ SFCP
CHLORIDE	~~At least once per 72 hours~~ SFCP
FLUORIDE	~~At least once per 72 hours~~ SFCP

*Not required with $T_{avg} \leq 250°F$

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

INSERT 1

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	~~At least once per 72 hours~~	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	~~1 per 14 days~~	1
3. Radiochemical for $\bar{E}$ Determination	~~1 per 6 months~~*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 µCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ µCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

#Until the specific activity of the primary coolant system is restored within its limits.

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

Insert 1

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

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

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

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

c. A maximum auxiliary spray water temperature differential of 625°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

Insert 1

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits ~~at least once per 30 minutes~~ during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit ~~at least once per 12 hours~~ during auxiliary spray operation.

Insert 1

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each RHR relief valve shall be demonstrated OPERABLE by:

a. Verifying the RHR relief valve isolation valves (8701A, 8701B, 8702A, and 8702B) are open ~~at least once per 72 hours~~ when the RHR relief valve is being used for overpressure protection.

INSERT 1

b. Testing pursuant to Specification 4.0.5.

c. Verification of the RHR relief valve setpoint of at least one RHR relief valve, ~~at least once per 18 months~~ on a rotating basis.

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

4.4.9.3.3 At least two charging pumps shall be verified incapable of injecting into the RCS ~~at least once per 31 days~~, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.

Insert 1

* Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, verify these valves open ~~at least once per 31 days~~.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 7489 and 7685 gallons,

c. A boron concentration of between 2200 and 2500 ppm, and

d. A nitrogen cover-pressure of between 600 and 656 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

INSERT 1

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. ~~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 accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

INSERT 1

b. ~~At least once per 31 days~~ and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution.

c. ~~At least once per 31 days~~ when the RCS pressure is above 2000 psig by verifying that the isolation valve operator breaker opened at the motor control center and locked in the open position.

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

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

2. Upon receipt of a safety injection test signal.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. ~~At least once per 12 hours~~ by verifying that the following valves are in the indicated positions with power to the valve operators removed:

	Valve Number	Valve Function	Valve Position
1.	8884	HHSI Hot Leg Injection	Closed
2.	8886	HHSI Hot Leg Injection	Closed
3.	8888A	LHSI Cold Leg Injection	Open
4.	8888B	LHSI Cold Leg Injection	Open
5.	8889	LHSI Hot Leg Injection	Closed
6.	8701A	RHR Inlet	Closed
7.	8701B	RHR Inlet	Closed
8.	8702A	RHR Inlet	Closed
9.	8702B	RHR Inlet	Closed
10.	8133A	Charging/HHSI Cross-Connect	Open
11.	8133B	Charging/HHSI Cross-Connect	Open
12.	8106	Charging Mini-Flow Header Isolation	Open

Insert 1

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

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

2. Verify ECCS locations susceptible to gas accumulation are sufficiently filled with water.

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

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

2. Of the areas affected with the reactor building at the completion of each reactor building entry when CONTAINMENT INTEGRITY is established.

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

1. Verifying automatic interlock action of the RHR system from the Reactor Coolant System by ensuring that, with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig, the interlocks prevent the valves from being opened.

* Not required to be met for system vent flow paths opened under administrative control.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection actuation and containment sump recirculation test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

a) Centrifugal charging pump

b) Residual heat removal pump

INSERT 1

f. By verifying each ECCS pump's developed head at the test flow point for that pump is greater than or equal to the required developed head in accordance with Specification 4.0.5.

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

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

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

HPSI System Valve Number	
a.	8996A
b.	8996B
c.	8996C
d.	8994A
e.	8994B
f.	8994C
g.	8989A
h.	8989B
i.	8989C
j.	8991A
k.	8991B
l.	8991C

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

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

4.5.3.2 All charging pumps except the above required OPERABLE pumps, shall be demonstrated inoperable ~~at least once per 31 days~~ whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F by verifying that the motor circuit breakers have been secured in the open position.

INSERT 1

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A minimum contained borated water volume of 453,800 gallons,

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

c. A minimum water temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

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

Insert 1

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

2. Verifying the boron concentration of the water.

b. ~~At least once per 24 hours~~ by verifying the RWST temperature when the outside air temperature is less than 40°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

INSERT 1

a. ~~At least once per 31 days~~ by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open under administrative control as permitted by Specification 3.6.4.

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

c. Deleted.

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

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each reactor building air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates in accordance with the Containment Leakage Rate Testing Program.

INSERT 1

b. Deleted.

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

d. Deleted.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Reactor building internal pressure shall be maintained between -0.1 and 1.5 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The reactor building internal pressure shall be determined to be within the limits ~~at least once per 12 hours~~.

INSERT 1

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at or above the following locations and shall be determined ~~at least once per 24 hours~~: INSERT 1

a. Elevation 412' - 3 locations

b. Elevation 436' - 3 locations

c. Elevation 463' - 3 locations

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 36-inch containment purge supply and exhaust isolation valve shall be sealed closed.

b. The 6-inch containment purge supply and exhaust isolation valves may be open for less than or equal to 1000 hours per 365 days.

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

ACTION:

a. With a 36-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed close, close and/or seal close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a 6-inch containment purge supply and/or exhaust isolation valve(s) open for more than 1000 hours per 365 days, close the open 6-inch valve(s) or isolate the penetration within 4 hours otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Surveillance Requirements 4.6.1.7.3, restore the inoperable valve(s) to OPERABLE status within 24 hours; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 36-inch containment purge supply and exhaust isolation valve shall be verified to be:

INSERT 1

a. Closed ~~at least once per 24 hours.~~

b. Sealed closed ~~at least once per 31 days.~~

4.6.1.7.2 The cumulative time that the 6-inch purge supply and exhaust isolation valves have been open during the past 365 days shall be determined ~~at least once per 7 days.~~

INSERT 1

4.6.1.7.3 ~~At least once per 30 months~~ each containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

REACTOR BUILDING SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent reactor building spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and automatically transferring suction to the spray sump.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each reactor building spray system shall be demonstrated OPERABLE:

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

Insert 1

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*, and

2. Verifying Containment Spray locations susceptible to gas accumulation are sufficiently filled with water.

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 195 psig when tested pursuant to Specification 4.0.5.

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on each of the following test signals a Phase 'A', Reactor Building Spray Actuation, and Containment Sump Recirculation.

2. Verifying that each spray pump starts automatically on a Reactor Building Spray Actuation test signal.

d. At least once per 10 years by performing an air or smoke or equivalent flow test through each spray header and verifying each spray nozzle is unobstructed.

* Not required to be met for system vent flow paths opened under administrative control.

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with:

a. A spray additive tank containing a volume of between 3140 and 3230 gallons of between 20.0 and 22.0 percent by weight NaOH solution, and

b. A flow path capable of adding NaOH solution from the spray additive tank to the suction of each reactor building spray pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

INSERT 1

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

b. ~~At least once per 6 months~~ by:

1. Verifying the contained solution volume in the tank, and
2. Verifying the concentration of the NaOH solution by chemical analysis.

c. ~~At least once per 18 months~~ during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Phase 'A' signal.

d. ~~At least once per 5 years~~ by verifying each solution flow rate from the following drain connections in the spray additive system:

1. NaOH Tank to Loop A $\geq$ 15 gpm
2. NaOH Tank to Loop B $\geq$ 15 gpm

CONTAINMENT SYSTEMS

REACTOR BUILDING COOLING SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.6.2.3 Two independent groups of reactor building cooling units shall be OPERABLE with at least one of two cooling units OPERABLE in slow speed in each group.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one group of the above required reactor building cooling units inoperable and both reactor building spray systems OPERABLE, restore the inoperable group of cooling units to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required reactor building cooling units inoperable, and both reactor building spray systems OPERABLE, restore at least one group of cooling units to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required reactor building cooling units inoperable and one reactor building 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 COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of reactor building cooling units shall be demonstrated OPERABLE:

INSERT 1

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

1. Starting each cooling unit group from the control room, and verifying that each cooling unit group operates for at least 15 minutes in the slow speed mode.

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

1. Verifying that each fan group starts automatically on a safety injection test signal.

2. Verifying a cooling water flow rate of greater than or equal to 2,000 gpm to each cooling unit group.

SURVEILLANCE REQUIREMENTS (Continued)

Insert 1

3. ~~At least once per 18 months~~ during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a simulated SI test signal or on an ESFLS, as applicable.

4. ~~At least once per 18 months~~, by verifying that each service water system booster pump starts automatically on a safety injection signal.

CONTAINMENT SYSTEMS

3/4.6.3 PARTICULATE IODINE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3 Two independent groups of HEPA filter banks (associated with the OPERABLE reactor building cooling units of Specification 3.6.2.3) with at least one filter bank in each group, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one group of HEPA filter banks OPERABLE, restore one of the inoperable banks in the other group to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3 The two groups of HEPA filter banks shall be demonstrated OPERABLE:

INSERT 1

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

b. By performing required filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

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

   1. Verifying that the filter bypass damper can be opened by operator action.

   2. Verifying that the filter bypass damper closes on a Safety Injection Test Signal.

CONTAINMENT SYSTEMS

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

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

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

The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test and verification of isolation time.

INSERT 1

4.6.4.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE ~~AT LEAST ONCE PER 18 MONTHS~~ BY:

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

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

c. Verifying that on a Reactor Building Purge and Exhaust isolation test signal, each Purge and Exhaust valve actuates to its isolation position.

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

PLANT SYSTEMS

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator emergency feedwater pumps and flow paths shall be OPERABLE with:

a. Two motor-driven emergency feedwater pumps, each capable of being powered from separate emergency busses, and

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one emergency feedwater pump inoperable, restore the required emergency feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two emergency feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three emergency feedwater pumps inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater pump shall be demonstrated OPERABLE:

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

Insert 1

1. Not used

2. Not used

3. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying that each automatic valve in the flow path from the condensate storage tank to the steam generators is in the fully open position whenever the emergency feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER.

5. Verifying that valves 1010-EF and 1007-EF are locked in the open position.

b. ~~At least once per 3 months~~ by verifying that the check valve in the instrument air supply line to the six emergency feedwater control valve air accumulators closes when the normal instrument air supply is not available.

Insert 1

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

1. Each emergency feed pump starts as designed automatically upon receipt of an emergency feedwater actuation test signal.

2. The six emergency feedwater control valves can be closed and held closed for three hours with air from the accumulators when the normal instrument air supply is not available.

3. The turbine driven emergency feedwater pump can be manually stopped from the main control board by closing the steam supply valve with air from the accumulator when the normal instrument air supply is not available.

4. Each automatic valve in the flow path actuates to its correct position on receipt of an emergency feedwater actuation test signal.

d. In accordance with the Inservice Testing Program as required by Specification 4.0.5 by verifying:

1. The developed head of each emergency feedwater pump at the flow test point is greater than or equal to the required developed head. Notes: 1) Not required to be performed for the turbine driven emergency feedwater pump until secondary steam supply pressure is greater than 865 psig. 2) The provisions of Specification 4.0.4 are not applicable for the turbine driven emergency feedwater pump.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained volume of at least 179,850 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the service water system as a backup supply to the emergency feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.7.1.3.1 The 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.

4.7.1.3.2 The service water system shall be demonstrated OPERABLE ~~at least once per 12 hours~~ by verifying service water system pressure whenever the service water system is the supply source for the emergency feedwater pumps.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

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

INSERT 1

PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of the primary coolant and the steam generator shells shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

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

SURVEILLANCE REQUIREMENTS

INSERT 1

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig ~~at least once per hour~~ when the temperature of either the primary coolant or the steam generator shell is less than 70°F.

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

INSERT 1

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

PLANT SYSTEMS

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

INSERT 1

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

PLANT SYSTEMS

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The service water pond (ultimate heat sink) shall be OPERABLE with:

a. A minimum water level at or above elevation 416.5 Mean Sea Level, USGS datum, and

b. A water temperature of less than or equal to 90.5°F at the discharge of the service water pumps.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

INSERT 1

4.7.5 The service water pond shall be determined OPERABLE ~~at least once per 24 hours~~ by verifying the water temperature and water level to be within their limits.

PLANT SYSTEMS

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS)

LIMITING CONDITION FOR OPERATION

3.7.6 Two CREFS trains shall be OPERABLE.*

APPLICABILITY: ALL MODES

ACTION:

a. MODES 1, 2, 3 and 4:

1. With one CREFS train inoperable for reasons other than ACTION 3.7.6.a.2, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With one or more CREFS trains inoperable due to an inoperable control room envelope (CRE) boundary, immediately initiate action to implement mitigating actions and verify within 24 hours that the mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits and restore CRE boundary to OPERABLE status within 90 days. Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. With both CREFS trains inoperable for reasons other than ACTION 3.7.6.a.2, immediately enter LCO 3.0.3.

b. MODES 5 and 6:

1. With one CREFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable train to OPERABLE status within 7 days, or immediately place the OPERABLE CREFS train in the emergency mode of operation or immediately suspend movement of irradiated fuel assemblies.

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

3. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6 Each CREFS train shall be demonstrated OPERABLE:

INSERT 1

a. ~~At least once per 12 hours~~ by verifying that the control room air temperature is less than or equal to 85°F.

b. ~~At least once per 31 days~~ by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the CREFS train operates for at least 15 minutes.

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

* The control room envelope (CRE) boundary may be opened intermittently under administrative control.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

INSERT 1

d. ~~At least once per 18 months~~ by verifying that on a simulated SI or high radiation test signal, each CREFS train automatically switches into an emergency mode of operation with flow through the HEPA filters and charcoal adsorber banks.

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

PLANT SYSTEMS

3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.8 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material, shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

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

1. Decontaminate and repair the sealed source, or

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

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

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

a. Sources in use - ~~At least once per six months~~ INSERT 1 for all sealed sources containing radioactive materials:

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

2. In any form other than gas.

PLANT SYSTEMS

3/4.7.9 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.9 The temperature of each area shown in Table 3.7-7 shall be maintained below the limits indicated in Table 3.7-7.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7-7:

a. For more than eight hours, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to below its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.9 The temperature in each of the areas of Table 3.7-7 shall be determined to be within its limit ~~at least once per 12 hours.~~

INSERT 1

PLANT SYSTEMS

3/4.7.10 WATER LEVEL-SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

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

SURVEILLANCE REQUIREMENTS

4.7.10 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.

INSERT 1

PLANT SYSTEMS

3/4.7.13 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.13 The boron concentration in the spent fuel pool, the fuel transfer canal, and the cask loading pit shall be maintained at a boron concentration greater than or equal to 500 ppm.

APPLICABILITY: Whenever new or irradiated fuel is being moved (non-refueling movement) in the spent fuel pool, fuel transfer canal, or cask loading pit.

ACTION:

With the requirements of the above not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel pool, the fuel transfer canal, and the cask loading pit until the boron concentration in the area where fuel is being moved shall be verified greater than or equal to 500 ppm.

SURVEILLANCE REQUIREMENTS

4.7.13 The boron concentration of the spent fuel pool, fuel transfer canal, or cask loading pit shall be determined by chemical analysis ~~at least once per 72 hours~~ when moving new or irradiated fuel in the spent fuel pool, transfer canal, or cask loading pit.

INSERT 1

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. With two of the required offsite A. C. circuits inoperable:

1. Demonstrate the OPERABILITY of the two EDG's by sequentially performing Surveillance Requirement 4.8.1.1.2.a.3 on both within 8 hours, unless the EDG's are already operating, and

2. Restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours.

3. Following restoration of one offsite source, follow Action Statement a. with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable offsite A.C. circuit.

e. With two of the above required EDG's inoperable:

1. Demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1 within one hour and at least once per 8 hours thereafter, and

2. Restore one of the inoperable EDG's to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. Following restoration of one EDG, follow Action Statement b. with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable diesel generator.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE ~~at least once per 7 days~~ by verifying correct breaker alignment and indication of power availability for each Class 1E bus and its preferred offsite power source.

INSERT 1

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each EDG shall be demonstrated OPERABLE:

INSERT 1

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

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

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

3. Verifying the diesel generator can start* and accelerate to synchronous speed (504 rpm) with generator voltage and frequency at 7200 ± 720 volts and 60 ± 1.2 Hz.

4. Verifying the generator is synchronized, gradually loaded* to an indicated 4150-4250 kW** and operates for at least 60 minutes.

b. ~~At least once per 31 days~~ and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by removing accumulated water from the day tank.

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

d. By sampling new fuel oil based on the applicable ASTM standard prior to addition to storage tanks and:

1. By verifying based on the tests specified in the applicable ASTM standard prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification;

---

* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

c) A flash point equal to or greater than 125°F; and

d) A clear and bright appearance when tested based on the applicable ASTM standard.

2. By verifying within 30 days of obtaining the sample that the specified properties are met when tested based on the applicable ASTM standard.

e. ~~At least once every 31 days~~ by obtaining a sample of fuel oil based on the applicable ASTM standard, and verifying that total contamination is less than 10 mg/liter when checked based on the applicable ASTM standard.

INSERT 1

f. ~~At least once per 184 days~~ by:

1. Verify each EDG starts from standby conditions and:

   a) In less than or equal to 10 seconds, achieves a voltage greater than 6480 volts (7200 - 720 volts) and a frequency greater than 58.8 Hz (60 - 1.2 Hz).

   b) Achieve a steady state voltage greater than 6480 volts but less than 7920 volts and a steady state frequency greater than 58.8 Hz but less than 61.2 Hz.

   The EDG shall be started for this test by using one of the following signals:

   a) Simulated loss of offsite power by itself.
   b) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
   c) An ESF actuation test signal by itself.
   d) Simulated degraded offsite power by itself.
   e) Manual.

2. The generator shall be manually synchronized, loaded to an indicated 4150-4250 kW** in less than or equal to 60 seconds, and operate for at least 60 minutes.

g. ~~At least once every 18 months~~ by:

1. Deleted

2. Verifying that on rejection of a load of greater than or equal to 729 kW, the voltage and frequency are maintained at 7200 ± 720 volts and frequency at 60 ± 1.2 Hz.

3. Verifying the generator capability to reject a load of 4250 kW without tripping. The generator voltage shall not exceed 7920 volts during and following the load rejection.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer, and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization of these loads, the steady-state voltage and frequency shall be maintained at 7200 ± 720 volts and 60 ± 1.2 Hz.

INSERT 1

h. ~~At least once per 10 years~~ or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 504 rpm in less than or equal to 10 seconds.

~~i. At least once per 10 years by:~~

~~1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or its equivalent, and~~

~~2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III subsection ND of the ASME Code in accordance with Specification 4.0.5.~~

i. In accordance with the Surveillance Frequency Control Program, by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or its equivalent.

j. At least oncer per 10 years, by performing a pressure test of those portions of the diesel fuel oil system designed to Section III subsection ND of the ASME Code in accordance with Specificaiton 4.0.5.

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

a. 125-volt Battery bank No. 1A and its associated full capacity charger.

b. 125-volt Battery bank No. 1B and its associated full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

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

INSERT 1

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

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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

Insert 1

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

2. There is no visible corrosion at either terminals or connectors, or the battery connection resistance is less than or equal to the individual connection resistance for the connection types listed below or total battery resistance is less than or equal to 2890 μΩ:

Maximum Individual Battery Connection Resistances		
Connection Type	Number of Connections	Individual Connection Resistance (μΩ)
Inter-cell	56	45
Jumper	3	100
Terminal Plate	2	35

, and

3. The average electrolyte temperature of 10 of the connected cells is ≥ 60°F.

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

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

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3. The battery connection resistance is less than or equal to the individual connection resistance for the connection types listed below or total battery resistance is less than or equal to 2890 μΩ:

Insert 1

Maximum Individual Battery Connection Resistances		
Connection Type	Number of Connections	Individual Connection Resistance (μΩ)
Inter-cell	56	45
Jumper	3	100
Terminal Plate	2	35

, and

4. The battery charger will supply at least 300 amperes at 132 volts for at least 8 hours.

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

e. ~~At least once per 60 months~~, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1.d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

ELECTRICAL POWER SYSTEMS

ACTION: (Continued)

d. With one D.C. bus not energized from its associated Battery Bank, re-energize the D.C. bus from its associated Battery Bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

INSERT 1

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

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. Emergency Busses consisting of two 7200 volt and three 480 volt A.C. Emergency Busses.

b. Three 120 volt A.C. Vital Busses energized from their associated inverters connected to their respective D.C. Busses.

c. One 125 volt D.C. Bus energized from its associated battery bank.

APPLICABITLITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, and initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

INSERT 1

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

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 For each containment penetration provided with a penetration conductor overcurrent protective device(s), each device(s) shall be operable.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

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

SURVEILLANCE REQUIREMENTS

4.8.4.1 Protective devices required to be operable as containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE.

a. ~~At least once per 18 months:~~ INSERT 1

1. By verifying that the medium voltage (7.2 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

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

(b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

INSERT 1

b. ~~At least once per 60 months~~ by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

ELECTRICAL POWER SYSTEMS

CIRCUIT PROTECTION DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.3 Circuit breakers for non-Class 1E cables located in trays which do not have cable tray covers and which provide protection for cables that if faulted could cause failure in both adjacent, redundant Class 1E cables shall be OPERABLE.

APPLICABILITY: All modes

ACTION:

a. With one or more of the above required non-Class 1E circuit breaker(s) inoperable, within 72 hours, either:

   1. Restore the circuit breaker(s) to OPERABLE status; or
   2. De-energize the circuit breaker(s); or
   3. Establish a one (1) hour roving fire watch for those areas in which redundant systems or components could be damaged.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above required circuit breakers shall be demonstrated OPERABLE.

a. ~~At least once per eighteen (18) months:~~ INSERT 1

   1. By verifying that the medium voltage (7.2 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers and performing the following:

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

      (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

      (c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing a representative sample of at least ten percent (10%) of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breaker's nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least ten percent (10%) of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. ~~At least once per sixty (60) months~~ by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

INSERT 1

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2000 ppm.

APPLICABILITY: MODE 6* with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

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 greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

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 full length control rod 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 system and the refueling canal shall be determined by chemical analysis ~~at least once per 72 hours.~~

4.9.1.3 The following valves shall be verified locked closed** ~~at least once per 72 hours:~~ 8430, 8454, 8441 and 8439.

INSERT 1

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

**Valves may be opened under administrative control to add borated makeup.

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6.

ACTION:

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

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the reactor coolant system at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

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

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

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST ~~at least once per 7 days.~~

INSERT 1

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical a period of time within the acceptable domain of Figure 3.9-1, but not less than 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, immediately suspend all movement of irradiated fuel in the reactor pressure vessel. With the reactor subcritical for greater than 72 hours but not within the acceptable domain of Figure 3.9-1, immediately suspend movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3.1 The reactor shall be determined to have been subcritical for a period of time within the acceptable domain of Figure 3.9-1 by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

INSERT 1

4.9.3.2 Prior to moving irradiated fuel from the reactor pressure vessel, and ~~at least once every 12 hours~~ during movement of irradiated fuel, verify the CCW temperature at the inlet to the Spent Fuel Pool Cooling System heat exchanger is within the acceptable domain of Figure 3.9-1.

3/4.9.5 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.

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 ~~at least once per 12 hours~~ during CORE ALTERATIONS.

INSERT 1

REFUELING OPERATIONS

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.7.1.1 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm ~~at least once per 12 hours~~.

4.9.7.1.2 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water ~~at least once per 31~~ days.

Insert 1

* The residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

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

ACTION:

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

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.7.2.1 At least one residual heat removal loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm ~~at least once per 12 hours.~~

4.9.7.2.2 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water ~~at least once per 31 days.~~

Insert 1

* Prior to initial criticality the residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

3/4.9.9 WATER LEVEL - REFUELING CAVITY AND FUEL TRANSFER CANAL

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel or the refueling cavity when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

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

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 ~~at least once per 24 hours~~ thereafter during movement of fuel assemblies or control rods.

INSERT 1

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

INSERT 1

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

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

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODE 1

ACTION:

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

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

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

INSERT 1

4.10.2.2 The Surveillance Requirements of the below listed Specifications (a. and b.) shall be performed ~~at least once per 12 hours~~ during PHYSICS TESTS:

a. Either Specifications 4.2.2.2 or 4.2.2.4 and Specification 4.2.2.5.

b. Specification 4.2.3.2.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

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

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

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

INSERT 1

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

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F ~~at least once per 30 minutes~~ during PHYSICS TESTS.

INSERT 1

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

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

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

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set less than or equal to 25% of RATED THERMAL POWER.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

INSERT 1

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

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating start up and PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

With the position indication system inoperable or with more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS INSERT 1

4.10.5 The above required rod position indication systems shall be determined to be OPERABLE within 24 hours prior to the start of and ~~at least once per 24 hours thereafter~~ during rod drop time measurements by verifying the demand position indication system and the rod position indication systems agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

* This requirement is not applicable during the initial calibration of the rod position indication system provided (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.1 Deleted by Amendment 104.
3.11.1.2 Deleted by Amendment 104.
3.11.1.3 Deleted by Amendment 104.

3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Condensate Storage Tank
b. Outside Temporary Storage Tank

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1 Deleted by Amendment 104.
4.11.1.2 Deleted by Amendment 104.
4.11.1.3 Deleted by Amendment 104.

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

Insert 1

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 131,000 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.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit ~~at least one per 24 hours~~ when radioactive materials are being added to the tank.

Insert 1

n. Snubber Testing Program

This program conforms to the examination, testing and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

1) This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.

2) The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b) subject to limitations and modifications listed in 10 CFR 50.55a(b) and subject to Commission approval.

3) The program shall, as allowed by 10 CFR 50.55a(b)(3)(v), meet Subsection ISTA, "General Requirements," and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants," or meet authorized alternatives pursuant to 10 CFR 50.55a(a)(3).

4) The 120-month program updates shall be made in accordance with 10 CFR 50.55a(g)(4), 10 CFR 50.55a(g)(3)(v) and 10 CFR 50.55a(b) (including 10 CFR 50.55a(b)(3)(v)) subject to the limitations and modifications listed therein.

INSERT 2

ATTACHMENT 4

<u>REVISED (CLEAN) TS PAGES</u>

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.
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.

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.77% delta k/k for 3 loop operation.

APPLICABILITY: MODES 1, and 2*.

ACTION:

With the SHUTDOWN MARGIN less than 1.77% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be demonstrated to be greater than or equal to 1.77% delta k/k:

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

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, in accordance with the Surveillance Frequency Control Program by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Surveillance Requirement 4.1.1.1.2 with the control banks at the maximum insertion limit of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k in accordance with the Surveillance Frequency Control Program. This comparison shall consider at least the following factors.

1. Reactor Coolant System boron concentration,
2. Control rod position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy production,
5. Xenon concentration, and
6. Samarium.

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 - MODES 3, 4 AND 5

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal the limits shown in Figure 3.1-3.

APPLICABILITY: MODES 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than the required value, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be demonstrated to be greater than or equal to the required value:

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

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

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

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid tanks via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1.1 At least one of the above required flow paths shall be demonstrated OPERABLE 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.

4.1.2.1.2 Demonstrate operability of the required charging pump per Surveillance 4.5.2.f

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. The flow path from the boric acid tanks via a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System.

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

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2 percent delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. 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.

b. In accordance with the Surveillance Frequency Control Program by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 30 gpm to the Reactor Coolant System.

Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300° F.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum contained borated water volume of 2700 gallons,
2. Between 7000 and 7700 ppm of boron, and
3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 51,500 gallons,
2. A minimum boron concentration of 2300 ppm, and
3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

a. In accordance with the Surveillance Frequency Control Program by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume, and

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

b. In accordance with the Surveillance Frequency Control Program by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 40° F.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

1. Verifying the boron concentration in the water,

2. Verifying the contained borated water volume of the water source, and

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

b. In accordance with the Surveillance Frequency Control Program by verifying the RWST temperature when the outside air temperature is less than 40°F.

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

   a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

   b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

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

   d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions in accordance with the Surveillance Frequency Control Program except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction in accordance with the Surveillance Frequency Control Program.

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The shutdown and control rod position indication system and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 12 steps in accordance with the Surveillance Frequency Control Program except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system at least once per 4 hours.

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of an ANALOG CHANNEL OPERATIONAL TEST in accordance with the Surveillance Frequency Control Program.

* With the reactor trip system breakers in the closed position.
# See Special Test Exception 3.10.5.

LIMITING CONDITION FOR OPERATION

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

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

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. In accordance with the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1 * and 2 * #

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Restore the rod to within the limit specified in the COLR, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR.

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

b. In accordance with the Surveillance Frequency Control Program thereafter.

* See Special Test Exceptions 3.10.2 and 3.10.3.
# With K_{eff} greater than or equal to 1.0

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR) figure entitled Rod Group Insertion Limits versus Thermal Power For Three Loop Operation.

APPLICABILITY: MODES 1 * and 2 * #.

ACTION:

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits in accordance with the Surveillance Frequency Control Program except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

* See Special Test Exceptions 3.10.2 and 3.10.3

With K_{eff} greater than or equal to 1.0.

SURVEILLANCE REQUIREMENTS

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

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

4.2.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 When in Base Load operation, the target flux difference of each OPERABLE excore channel shall be determined by measurement in accordance with the Surveillance Frequency Control Program. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 When in Base Load operation, the target flux difference shall be updated in accordance with the Surveillance Frequency Control Program by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map

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

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

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

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

c. Satisfying the following relationship:

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

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

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing tolerances and measurement uncertainty as specified in the COLR, F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z) and W(z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

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

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined, * or

2. In accordance with the Surveillance Frequency Control Program, whichever occurs first.

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

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND} when the Power Distribution Monitoring System (PDMS) is inoperable; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS at any THERMAL POWER greater than APL^{ND}; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

c. Satisfying the following relationship:

$$F_Q^M(z) \le \frac{F_Q^{RTP} \times K(z)}{P \times W(z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing and measurement uncertainties as specified in the COLR. The F_Q limit is F_Q^{RTP}. P is the relative THERMAL POWER. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. F_Q^{RTP}, K(z) and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

d. Measuring $F_Q^M(z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering BASE LOAD operation after satisfying Section 4.2.2.3 unless a core power distribution measurement has been obtained in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to measurement, and

2. In accordance with the Surveillance Frequency Control Program.

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through a core power distribution measurement and RCS total flow rate comparison, to be within the region of acceptable operation specified in the COLR prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation specified in the COLR.

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. In accordance with the Surveillance Frequency Control Program.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation specified in the COLR In accordance with the Surveillance Frequency Control Program, when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program,

4.2.3.5 The RCS total flow rate shall be determined by heat balance measurement at ≥90% RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Calculating the ratio in accordance with the Surveillance Frequency Control Program when the alarm is OPERABLE.

b. Calculating the ratio in accordance with the Surveillance Frequency Control Program during steady state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 percent RATED THERMAL POWER with one Power Range Channel inoperable in accordance with the Surveillance Frequency Control Program by using the PDMS or movable incore detectors to confirm that the normalized symmetric power distribution is consistent with the indicated QUADRANT POWER TILT RATIO. The incore detector monitoring shall be done with 2 sets of 4 symmetric thimbles or a full incore flux map.

POWER DISTRIBUTION LIMITS

3/4 2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}*

b. Pressurizer Pressure

APPLICABILITY: MODE 1*

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters of Table 3.2-1 shall be verified to be within their limits in accordance with the Surveillance Frequency Control Program.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor trip system instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by performance of the reactor trip system instrumentation surveillance requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be verified to be within its limit in accordance with the Surveillance Frequency Control Program.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A	N.A.	SFCP (11)	N.A.	1, 2, 3* ,4*, 5*
2. Power Range, Neutron Flux						
High Setpoint	SFCP	SFCP(2, 4), SFCP (3, 4), SFCP 4, 6), SFCP (4, 5)	SFCP	N.A.	N.A.	1, 2
Low Setpoint	SFCP	SFCP (4)	S/U(18), (16)	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux High Positive Rate	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1, 2
4. Deleted						
5. Intermediate Range, Neutron Flux	SFCP	SFCP (4)	S/U(18), (16)	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	SFCP	SFCP (4)	S/U(18), (17), (9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
8. Overpower ΔT	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	SFCP	SFCP	SFCP	N.A.	N.A.	1
10. Pressurizer Pressure--High	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	SFCP	SFCP	SFCP	N.A.	N.A.	1
12. Loss of Flow	SFCP	SFCP	SFCP	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level–Low-Low	SFCP	SFCP	SFCP	N.A.	N.A.	1,2
14. Steam Generator Water Level - Low Coincident with Steam/ Feedwater Flow Mismatch	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.	SFCP	N.A.	SFCP	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	SFCP	N.A.	SFCP	N.A.	1
17. Turbine Trip						
A. Low Fluid Oil Pressure	N.A.	SFCP	N.A.	(1, 8, 10)	N.A.	1
B. Turbine Stop Valve Closure	N.A.	SFCP	N.A.	(1, 8, 10)	N.A.	1
19. Reactor Trip System Interlocks						
A. Intermediate Range Neutron Flux, P-6	N.A.	SFCP (4)	SFCP	N.A.	N.A.	2##
B. Low Power Reactor Trips Block, P-7	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1
C. Power Range Neutron Flux, P-8	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT		CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
D	Low Setpoint Power Range Neutron Flux, P-10	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1,2
E.	Turbine Impulse Chamber Pressure, P-13	N.A.	SFCP	SFCP	N.A.	N.A.	1
F.	Low Power Range Neutron Flux, P-9	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1
20.	Reactor Trip Breaker	N.A.	N.A.	N.A.	(7, 12)	N.A.	1, 2, 3,* 4*, 5*
21.	Automatic Trip Logic	N.A.	N.A.	N.A.	N.A.	SFCP (15)	1, 2, 3*, 4*, 5*
22.	Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	(7, 13), SFCP (14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

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

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

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

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

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

(3) - Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

(5) - Detector plateau curves shall be obtained evaluated and compared to manufacturer's data. For the Power Range Neutron Flux Channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) - Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) - Each train shall be tested in accordance with the Surveillance Frequency Control Program.

(8) - DELETED

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

(10) - Setpoint verification is not required.

(11) - The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

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

(13) - Local manual shunt trip prior to placing breaker in service.

(14) - Automatic undervoltage trip.

(15) - Each train shall be tested in accordance with the Surveillance Frequency Control Program.

(16) 12 hours after reducing power below P-10 and in accordance with the Surveillance Frequency Control Program.

(17) 4 hours after reducing power below P-6 and 4 hours after entering MODE 3 from MODE 2 and in accordance with the Surveillance Frequency Control Program.

(18) If not performed in previous 184 days.

INSTRUMENTATION

3/4 3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint Column but more conservative than the value shown in the Allowable Value Column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value Column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to its OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

c. With an ESFAS instrumentation channel or interlock inoperable take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the engineered safety feature actuation system instrumentation surveillance requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified 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	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER								
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP (1)	SFCP (3)	1, 2, 3, 4
c. Reactor Building Pressure-High-1	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure--Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Line Pressure Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. REACTOR BUILDING SPRAY								
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP (1)	SFCP (3)	1, 2, 3, 4
c. Reactor Building Pressure-High-3	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continue)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. CONTAINMENT ISOLATION								
a. Phase "A" Isolation								
1) Manual	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						
3) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP1)	SFCP (3)	1, 2, 3, 4
b. Phase "B" Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP (1)	SFCP (3)	1, 2, 3, 4
2) Reactor Building Pressure-High-3	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Purge and Exhaust Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP (1)	SFCP (2,3)	1, 2, 3, 4
2) Containment Radioactivity-High	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
3) Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. STEAM LINE ISOLATION								
a. Manual	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP (1)	SFCP (3)	1, 2, 3
c. Reactor Building Pressure-High-2	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION								
a. Steam Generator Water Level--High-High	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2
b. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP (1)	SFCP (3)	1, 2
6. EMERGENCY FEEDWATER								
a. Manual	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP (1)	SFCP (3)	1, 2, 3
c. Steam Generator Water Level--Low-Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
EMERGENCY FEEDWATER (Continued)								
d. Undervoltage - Both ESF Busses	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						
f. Undervoltage - One ESF Bus	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2
h. Suction transfer on low pressure	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. LOSS OF POWER								
a. 7.2 kV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 7.2 kV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	SFCP	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
8. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP								
a. RWST level low-low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP (1)	SFCP (1)	SFCP (3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERA-TIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS								
a. Pressurizer Pressure, P-11	N.A.	SFCP	SFCP	N.A.	N.A	N.A.	N.A.	1, 2, 3
b. Low, Low Tavg, P-12	N.A.	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train shall be tested in accordance with the Surveillance Frequency Control Program.

(2) The 36 inch containment purge supply and exhaust isolation valves are sealed closed during Modes 1 through 4, as required by TS 3.6.1.7. With these valves sealed closed, their ability to open is defeated; therefore, they are excluded from the quarterly slave relay test.

(3) Slave Relay Testing will be conducted in accordance with the Surveillance Frequency Control Program for Westinghouse type AR relays and preferably during a refueling outage to preclude the risk of actuation. Replacement relays other than Westinghouse type AR or reconciled Cutler-Hammer relays will require further analysis and NRC approval to maintain the established frequency.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	SFCP	SFCP	SFCP	*
b. Deleted				
2. PROCESS MONITORS				
a. Deleted				
b. Containment				
i. Deleted				
ii. Particulate and Gaseous Activity - RCS Leakage Detection (RM-A2)	SFCP	SFCP	SFCP	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	SFCP	SFCP	SFCP	ALL MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	SFCP	SFCP	SFCP	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	SFCP	SFCP	SFCP	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	SFCP	SFCP	SFCP	1, 2, 3 & 4

* With fuel in the storage pool or building

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO using a full-core flux map per Specification 4.2.4.2, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(z)$.

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program, by normalizing each detector output when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(z)$.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Wind Speed Lower 10m	SFCP	SFCP
b. Wind Speed Upper 61m	SFCP	SFCP
2. Wind Direction		
a. Wind Direction Lower 10m	SFCP	SFCP
b. Wind Direction Upper 61m	SFCP	SFCP
3. Atmospheric Stability		
a. Delta T 1 10-61m	SFCP	SFCP
b. Delta T 2 10-40m	SFCP	SFCP

Elevations nominal above grade elevation

TABLE 4.3-6
REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Reactor Trip Breaker Indication	SFCP	N.A.
2.	Pressurizer Pressure	SFCP	SFCP
3.	Pressurizer Level	SFCP	SFCP
4.	Steam Generator Pressure	SFCP	SFCP
5.	Steam Generator Level	SFCP	SFCP
6.	Condensate Storage Tank Level	SFCP	SFCP
7.	Reactor Coolant System Hot Leg Temperature	SFCP	SFCP
8.	Reactor Coolant System Cold Leg Temperature	SFCP	SFCP
9.	Reactor Coolant System Pressure	SFCP	SFCP
10.	Pressurizer Relief Tank Level	SFCP	SFCP
11.	Reactor Building Temperature	SFCP	SFCP
12.	Boric Acid Tank Level	SFCP	SFCP

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown on Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 30 days or submit a Special Report within the following 14 days from the time the action is required. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to operable status.

b.1 With the number of OPERABLE Reactor Building radiation monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

i) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

ii) Submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

b.2 Deleted

b.3 With the number of OPERABLE accident monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performing a CHANNEL CHECK and a CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program. The Reactor Building Radiation Level Instrumentation CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for the range decades above 10R/hr and a single point calibration of the detector below 10R/hr with an installed or portable gamma source.

TABLE 4.3-9

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM				
a. Hydrogen Monitor	SFCP	SFCP (1)	SFCP	**
b. Oxygen Monitor	SFCP	SFCP (2)	SFCP	**

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACTION:

a. With one or more loose part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to 10 CFR 50.4 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK in accordance with the Surveillance Frequency Control Program,

b. An ANALOG CHANNEL OPERATIONAL TEST in accordance with the Surveillance Frequency Control Program, and

c. A CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION (Continued)

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

ACTION:

With any of the operability criteria listed in 3.3.3.11.a, 3.3.3.11.b, or 3.3.3.11.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system, satisfying the OPERABILITY requirements listed in Specification 3.3.3.2, to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11.1 The operability criteria listed in 3.3.3.11.a, 3.3.3.11.b, and 3.3.3.11.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

4.3.3.11.2 Calibration of the PDMS is required:

a. In accordance with the Surveillance Frequency Control Program when the minimum number and core coverage criteria as defined in 3.3.3.11.b.1 and 3.3.3.11.b.2 are satisfied, or |

b. In accordance with the Surveillance Frequency Control Program when only the minimum number criterion as defined in 3.3.3.11.b.3 is satisfied. |

3/4.4 REACTIVITY CONTROL SYSTEMS

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All Reactor Coolant loops shall be in operation,.

APPLICABILITY: MODES 1 and 2 *.

ACTION:

With less than the above required Reactor Coolant loops in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE 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 Exceptions 3.10.4.

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the Reactor Coolant loops listed below shall be OPERABLE and at least one of these Reactor Coolant loops shall be in operation.*

a. Reactor Coolant Loop A and its associated steam generator and Reactor Coolant Pump,

b. Reactor Coolant Loop B and its associated steam generator and Reactor Coolant Pump,

c. Reactor Coolant Loop C and its associated steam generator and Reactor Coolant Pump,

APPLICABILITY: MODE 3

ACTIONS:

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

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% of wide range indication 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.

SURVEILLANCE REQUIREMENTS

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

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% of wide range indication in accordance with the Surveillance Frequency Control Program.

4.4.1.3.3 At least one Reactor Coolant or RHR loop shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

4.4.1.3.4 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.*

* Not required to be performed until 12 hours after entering MODE 4.

COLD SHUTDOWN – LOOPS FILLED

LIMITING CONDITION FOR OPERATION

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

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

b. The secondary side water level of at least two steam generators shall be greater than 10 percent of wide range indication.

APPLICABILITY: MODE 5 with Reactor Coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE and/or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no residual heat removal 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 residual heat removal loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits in accordance with the Surveillance Frequency Control Program.

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

4.4.1.4.1.3 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

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

A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless 1) the pressurizer water volume is less than 1288 cubic feet and/or 2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

* The RHR pump 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.

COLD SHUTDOWN – LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with Reactor Coolant loops not filled.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.4.1.4.2.2 Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

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

* The RHR pump may be 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.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1288 cubic feet, (92% of indicated span) and at least two groups of pressurizer heaters each having a capacity of at least 125 kW.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current in accordance with the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION

ACTIONS: (Continued)

g. With three block valves inoperable:
   1) within 1 hour:
      a) restore the block valves to OPERABLE status, or
      b) place the associated PORVs in manual control and
   2) within the next 2 hours restore at least one of the three block valves to OPERABLE status and
   3) within the next 72 hours:
      a) restore at least two of the three block valves to OPERABLE status and
      b) ensure that the remaining inoperable block valve is closed and the power is removed;

   otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT STUTDOWN within the following 6 hours.

h. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by operating the valve through one complete cycle of full travel during MODES 3 or 4.

4.4.4.2 Each block valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by operating the valve through one complete cycle of full travel unless the block valve is closed with the power removed in order to meet the requirements of 3.4.4.b, 3.4.4.c, or 3.4.4.d.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

detection monitor, analyze grab samples of the containment atmosphere at least once per 12 hours and restore the required reactor building sump level monitor or the reactor building cooling unit condensate flow rate monitor to OPERABLE status within 7 days; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With the required reactor building atmosphere radioactivity monitor and the reactor building cooling unit condensate flow rate monitor inoperable, restore the required reactor building atmosphere radioactivity monitor or the reactor building air cooler condensate flow rate monitor to OPERABLE status within 30 days; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

f. With all required monitoring systems inoperable, enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Reactor building atmosphere particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Reactor building sump level-performance of CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program,

c. Reactor building atmosphere gaseous radioactivity monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION, AND ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified is Table 4.3-3,

d. Reactor building cooling unit condensate flow detector-performance of CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

---

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

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator,

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

e. 33 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. The leakage rate specified for each Reactor Coolant System Pressure Isolation Valve in Table 3.4-1 at a Reactor Coolant System pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any operational Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, primary-to-secondary leakage, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve Leakage greater than the limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the reactor building atmosphere (gaseous or particulate) radioactivity monitor in accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the reactor building sump inventory in accordance with the Surveillance Frequency Control Program.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig in accordance with the Surveillance Frequency Control Program days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program.[1] This requirement is not applicable to primary-to-secondary leakage.

e. Monitoring the reactor head flange leakoff system in accordance with the Surveillance Frequency Control Program.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit.

a. During startup in accordance with the Surveillance Frequency Control Program.

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

c. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve for valves denoted on Table 3.4-1 by an asterisk*.

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

4.4.6.2.3 Primary-to-secondary leakage shall be verified ≤ 150 gallons per day through any one steam generator in accordance with the Surveillance Frequency Control Program.[1]

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

TABLE 4.4-3

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETERS	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	SFCP
CHLORIDE	SFCP
FLUORIDE	SFCP

*Not required with $T_{avg} \leq 150^{\circ}F$

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

	TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1.	Gross Activity Determination	SFCP	1, 2, 3, 4
2.	Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	SFCP	1
3.	Radiochemical for $\overline{E}$ Determination	SFCP*	1
4.	Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 µCi/gram DOSE EQUIVALENT I-131 or 100/$\overline{E}$ µCi/gram, and	1#, 2#, 3#, 4#, 5#
		b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

\# Until the specific activity of the primary coolant system is restored within its limits.

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of0limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperatures shall be limited to:

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

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

c. A maximum auxiliary spray water temperature differential of 625°F

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit in accordance with the Surveillance Frequency Control Program during auxiliary spray operation.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each RHR relief valve shall be demonstrated OPERABLE by:

a. Verifying the RHR relief valve isolation valves (8701A, 8701B, 8702A, and 8702B) are open in accordance with the Surveillance Frequency Control Program when the RHR relief valve is being used for overpressure protection.

b. Testing pursuant the Specification 4.0.5.

c. Verification of the RHR relief valve setpoint of at least one RHR relief valve, in accordance with the Surveillance Frequency Control Program on a rotating basis.

4.4.9.3.2 The RCS vent shall be verified to be open in accordance with the Surveillance Frequency Control Program* when the vent is being used for overpressure protection.

4.4.9.3.3 At least two charging pumps shall be verified incapable of injecting into the RCS in accordance with the Surveillance Frequency Control Program, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.

* Except when the vent pathway is provided with a valve which is locked, sealed or otherwise secured in the open position, verify these valves open in accordance with the Surveillance Frequency Control Program.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 7489 and 7685 gallons,

c. A boron concentration of between 2200 and 2500 ppm, and

d. A nitrogen cover-pressure of between 600 and 656 psig.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and ion HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE.

a. In accordance with the Surveillance Frequency Control Program by:

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

   2. Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution.

c. In accordance with the Surveillance Frequency Control Program when the RCS pressure is above 2000 psig by verifying that the isolation valve operator breaker opened at the motor control center and locked in the open position.

d. In accordance with the Surveillance Frequency Control Program by verifying that each accumulator isolation valve opens automatically under each of the following conditions:

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

2. Upon receipt of a safety injection test signal.

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 valve operators removed:

	Valve Number	Valve Function	Valve Position
1.	8884	HHSI Hot Leg Injection	Closed
2.	8886	HHSI Hot Leg Injection	Closed
3.	8888A	LHSI Cold Leg Injection	Open
4.	8888B	LHSI Cold Leg Injection	Open
5.	8889	LHSI Hot Leg Injection	Closed
6.	8701A	RHR Inlet	Closed
7.	8701B	RHR Inlet	Closed
8.	8702A	RHR Inlet	Closed
9.	8702B	RHR Inlet	Closed
10.	8133A	Charging/HHSI Cross-Connect	Open
11.	8133B	Charging/HHSI Cross-Connect	Open
12.	8106	Charging Mini-Flow Header Isolation	Open

b. 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*, and

2. Verify ECCS locations susceptible to gas accumulation are sufficiently filled with water.

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

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

2. Of the areas affected with the reactor building at the completion of each reactor building entry when CONTAINMENT INTEGRITY is established.

d. In accordance with the Surveillance Frequency Control Program by:

1. Verifying automatic interlock action of the RHR system from the Reactor Coolant System by ensuring that, with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig, the interlocks prevent the valves from being opened.

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* Not required to be met for system vent flow paths opened under administrative control.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump component (trash, racks, screens, etc.) show no evident of structural distress or abnormal 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 a safety injection actuation and containment sump recirculation test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

a) Centrifugal charging pump

b) Residual heat removal pump

f. By verifying each ECCS pump's developed head at the test flow point for the pump is greater than or equal to the required developed head in accordance with Specification 4.0.5

g. By verifying the correct position of each mechanical position stope for the following ECCS throttle valves:

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

2. In accordance with the Surveillance Frequency Control Program.

HPSI System Valve Number	
a.	8886A
b.	8996B
c.	8996C
d.	8994A
e.	8994B
f.	8994C
g.	8989A
h.	8989B
i.	8989C
j.	8991A
k.	8991B
l.	8991C

SURVEILLANCE REQUIREMENTS

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

4.5.3.2 All charging pumps except the above required OPERABLE pumps, shall be demonstrated inoperable in accordance with the Surveillance Frequency Control Program whenever the temperature of one or more of the RCS Cold legs is less than or equal to 300°F by verifying that the motor circuit breakers have been secured in the open position.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A minimum contained borated water volume of 453,800 gallons,

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

c. A minimum water temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST 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 RWST temperature when the outside air temperature is less than 40°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1,2,3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. In accordance with the Surveillance Frequency Control Program by verifying that all penetrations * not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open under administrative control as permitted by Specification 3.6.4.

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

c. Deleted.

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

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each reactor building air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates in accordance with the Containment Leakage Rate Testing Program .

b. Deleted.

c. In accordance with the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

d. Deleted.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Reactor building internal pressure shall be maintained between -0.1 and 1.5 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The reactor building internal pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at or above the following locations and shall be determined in accordance with the Surveillance Frequency Control Program:

a. Elevation 412' - 3 locations

b. Elevation 436' - 3 locations

c. Elevation 463' - 3 locations

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 36-inch containment purge supply and exhaust isolation valve shall be sealed closed.

b. The 6-inch containment purge supply and exhaust isolation valves may be open for less than or equal to 1000 hours per 365 days.

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

ACTION:

a. With a 36-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed close, close and/or seal close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a 6-inch containment purge supply and/or exhaust isolation valve(s) open for more than 1000 hours per 365 days, close the open 6-inch valve(s) or isolate the penetration within 4 hours otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Surveillance Requirements 4.6.1.7.3, restore the inoperable valve(s) to OPERABLE status within 24 hours; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 36-inch containment purge supply and exhaust isolation valve shall be verified to be:

a. Closed in accordance with the Surveillance Frequency Control Program.

b. Sealed closed in accordance with the Surveillance Frequency Control Program.

4.6.1.7.2 The cumulative time that the 6-inch purge supply and exhaust isolation valves have been open during the past 365 days shall be determined in accordance with the Surveillance Frequency Control Program.

4.6.1.7.3 In accordance with the Surveillance Frequency Control Program each containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

REACTOR BUILDING SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent reactor building spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and automatically transferring suction to the spray sump.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each reactor building spray system 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*, and

2. Verifying Containment Spray locations susceptible to gas accumulation are sufficiently filled with water.

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 195 psig when tested pursuant to Specification 4.0.5.

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 each of the following test signals a Phase 'A', Reactor Building Spray Actuation, and Containment Sump Recirculation.

2. Verifying that each spray pump starts automatically on a Reactor Building Spray Actuation test signal.

d. At least once per 10 years by performing an air or smoke or equivalent flow test through each spray header and verifying each spray nozzle is unobstructed.

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* Not required to be met for system vent flow paths opened under administrative control.

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with:

a. A spray additive tank containing a volume of between 3140 and 3230 gallons of between 20.0 and 22.0 percent by weight NaOH solution, and

b. A flow path capable of adding NaOH solution from the spray additive tank to the suction of each reactor building spray pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

a. 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.

b. In accordance with the Surveillance Frequency Control Program by:

   1. Verifying the contained solution volume in the tank, and

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

c. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Phase 'A' signal.

d. In accordance with the Surveillance Frequency Control Program by verifying each solution flow rate from the following drain connections in the spray additive system:

   1. NaOH Tank to Loop A $\geq$ 15 gpm

   2. NaOH Tank to Loop B $\geq$ 15 gpm

REACTOR BUILDING COOLING SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.6.2.3 Two independent groups of reactor building cooling units shall be OPERABLE with at least one of two cooling units OPERABLE in slow speed in each group.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one group of the above required reactor building cooling units inoperable and both reactor building spray systems OPERABLE, restore the inoperable group of cooling units to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required reactor building cooling units inoperable, and both reactor building spray systems OPERABLE, restore at least one group of cooling units to OPERABLE status within 72 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required reactor building cooling units inoperable and one reactor building 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 COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of reactor building cooling units shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

    1. Starting each cooling unit group from the control room, and verifying that each cooling unit group operates for at least 15 minutes in the slow speed mode.

b. In accordance with the Surveillance Frequency Control Program by:

    1. Verifying that each fan group starts automatically on a safety injection test signal.

    2. Verifying a cooling water flow rate of greater than or equal to 2,000 gpm to each cooling unit group.

SURVEILLANCE REQUIREMENTS (Continued)

3. 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 a simulated SI test signal or on an ESFLS, as applicable.

4. In accordance with the Surveillance Frequency Control Program, by verifying that each service water system booster pump starts automatically on a safety injection signal.

3/4.6.3 PARTICULATE IODINE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3 Two independent groups of HEPA filter banks (associated with the OPERABLE reactor building cooling units of Specification 3.6.2.3) with at least one filter bank in each group, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one group of HEPA filter banks OPERABLE, restore one of the inoperable banks in the other group to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3 The two groups of HEPA filter banks shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and verifying that the system operates for at least 15 minutes. |

b. By performing required filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

c. In accordance with the Surveillance Frequency Control Program by: |

1. Verifying that the filter bypass damper can be opened by operator action.

2. Verifying that the filter bypass damper closes on a Safety Injection Test Signal.

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

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

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

The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test and verification of isolation time.

4.6.4.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE in accordance with the Surveillance Frequency Control Program by:

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

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

c. Verifying that on a Reactor Building Purge and Exhaust isolation test signal, each Purge and Exhaust valve actuates to its isolation position.

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

PLANT SYSTEMS

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator emergency feedwater pumps and flow paths shall be OPERABLE with:

a. Two motor-driven emergency feedwater pumps, each capable of being powered from separate emergency busses, and

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one emergency feedwater pump inoperable, restore the required emergency feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two emergency feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three emergency feedwater pumps inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump to OPERABLE status as soon as possible.

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. Not used

2. Not used

3. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

4. Verifying that each automatic valve in the flow path from the condensate storage tank to the steam generators is in the fully open position whenever the emergency feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER.

5. Verifying that valves 1010-EF and 1007-EF are locked in the open position.

b. In accordance with the Surveillance Frequency Control Program by verifying that the check valve in the instrument air supply line to the six emergency feedwater control valve air accumulators closes when the normal instrument air supply is not available.

c. In accordance with the Surveillance Frequency Control Program during shutdown by verifying that:

1. Each emergency feed pump starts as designed automatically upon receipt of an emergency feedwater actuation test signal.

2. The six emergency feedwater control valves can be closed and held closed for three hours with air from the accumulators when the normal instrument air supply is not available.

3. The turbine driven emergency feedwater pump can be manually stopped from the main control board by closing the steam supply valve with air from the accumulator when the normal instrument air supply is not available.

4. Each automatic valve in the flow path actuates to its correct position on receipt of an emergency feedwater actuation test signal.

d. In accordance with the Inservice Testing Program as required by Specification 4.0.5 by verifying:

1. The developed head of each emergency feedwater pump at the flow test point is greater than or equal to the required developed head. Notes: 1) Not required to be performed for the turbine driven emergency feedwater pump until secondary steam supply pressure is greater than 865 psig. 2) The provisions of Specification 4.0.4 are not applicable for the turbine driven emergency feedwater pump.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained volume of at least 179,850 gallons of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the service water system as a backup supply to the emergency feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The 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 service water system pressure whenever the service water system is the supply source for the emergency feedwater pumps.

TABLE 4.7-1

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 indicates iodine concentrations greater than 10% of the allowable limit.
	b) In accordance with the Surveillance Frequency Control Program, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of the primary coolant and the steam generator shells shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

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

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig in accordance with the Surveillance Frequency Control Program when the temperature of either the primary coolant or the steam generator shell is less than 70°F. |

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

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.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. 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.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The service water pond (ultimate heat sink) shall be OPERABLE with:

a. A minimum water level at or above elevation 416.5 Mean Sea Level, USGS datum, and

b. A water temperature of less than or equal to 90.5°F at the discharge of the service water pumps.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The service water pond shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the water temperature and water level to be within their limits.

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS)

LIMITING CONDITION FOR OPERATION

3.7.6 Two CREFS trains shall be OPERABLE.*

APPLICABILITY: ALL MODES

ACTION:

a. MODES 1, 2, 3 and 4:

1. With one CREFS train inoperable for reasons other than ACTION 3.7.6.a.2, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With one or more CREFS trains inoperable due to an inoperable control room envelope (CRE) boundary, immediately initiate action to implement mitigating actions and verify within 24 hours that the mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits and restore CRE boundary to OPERABLE status within 90 days. Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. With both CREFS trains inoperable for reasons other than ACTION 3.7.6.a.2, immediately enter LCO 3.0.3.

b. MODES 5 and 6:

1. With one CREFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable train to OPERABLE status within 7 days, or immediately place the OPERABLE CREFS train in the emergency mode of operation or immediately suspend movement of irradiated fuel assemblies.

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

3. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6 Each CREFS train shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the control room air temperature is less than or equal to 85°F. |

b. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the CREFS train operates for at least 15 minutes. |

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

* The control room envelope (CRE) boundary may be opened intermittently under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

d. In accordance with the Surveillance Frequency Control Program by verifying that on a simulated SI or high radiation test signal, each CREFS train automatically switches into an emergency mode of operation with flow through the HEPA filters and charcoal adsorber banks.

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

3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.8 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material, shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

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

   1. Decontaminate and repair the sealed source, or

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

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

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

a. Sources in use - In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive materials:

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

   2. In any form other than gas.

3/4.7.9 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.9 The temperature of each area shown in Table 3.7-7 shall be maintained below the limits indicated in Table 3.7-7.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7-7:

a. For more than eight hours, prepare and submit a Special Report to the Commission pursuant to 10 CFR 50.4 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to below its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.9 The temperature in each of the areas of Table 3.7-7 shall be determined to be within its limit in accordance with the Surveillance Frequency Control Program.

3/4.7.10 WATER LEVEL-SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

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

SURVEILLANCE REQUIREMENTS

4.7.10 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.

3/4.7.13 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.13 The boron concentration in the spent fuel pool, the fuel transfer canal, and the cask loading pit shall be maintained at a boron concentration greater than or equal to 500 ppm.

APPLICABILITY: Whenever new or irradiated fuel is being moved (non-refueling movement) in the spent fuel pool, fuel transfer canal, or cask loading pit.

ACTION:

With the requirements of the above not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel pool, the fuel transfer canal, and the cask loading pit until the boron concentration in the area where fuel is being moved shall be verified greater than or equal to 500 ppm.

SURVEILLANCE REQUIREMENTS

4.7.13 The boron concentration of the spent fuel pool, fuel transfer canal, or cask loading pit shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program when moving new or irradiated fuel in the spent fuel pool, transfer canal, or cask loading pit.

ACTION: (Continued)

d. With two of the required offsite A. C. Circuits inoperable:

1. Demonstrate the OPERABILITY of the two EDG's by sequentially performing Surveillance Requirement 4.8.1.1.2.a.3 on both within 8 hours, unless the EDG's are already operating, and

2. Restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours.

3. Following restoration of one offsite source, follow Action Statement a. with the time requirements of that Action Statement based on the time of initial loss of the remaining inoperable offsite A. C. circuit.

e. With two of the above required EDG's inoperable:

1. Demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirements 4.8.1.1.1 within one hour and at least once per 8 hours thereafter, and

2. Restore one of the inoperable EDG's to OPERABLE status within 2 hours or bin in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. Following restoration of one EDG, follow Action Statement b. wit the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable diesel generator.

SURVEILLANCE REQUIRMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indication of power availability for each Class 1E bus and its preferred offsite power source.

4.8.1.1.2 Each EDG shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

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

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

3. Verifying the diesel generator can start* and accelerate to synchronous speed (504 rpm) with generator steady state voltage greater than or equal to 6840 volts and less than or equal to 7445 volts and generator steady state frequency at $\pm$ 0.6 Hz.

4. Verifying the generator is synchronized, gradually loaded* to an indicated 4150-4250 kW** and operates for at least 60 minutes.

b. In accordance with the Surveillance Frequency Control Program and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by removing accumulated water from the day tank.

c. In accordance with the Surveillance Frequency Control Program by checking for and removing accumulated water from the fuel oil storage tanks.

d. By sampling new fuel oil based on the applicable ASTM standard prior to addition to storage tanks and:

1. By verifying based on the tests specified in the applicable ASTM standard prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification;

* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

c) A flash point equal to or greater than 125°F; and

d) A clear and bright appearance when tested based on the applicable ASTM standard.

2. By verifying within 30 days of obtaining the sample that the specified properties are met when tested based on the applicable ASTM standard.

e. In accordance with the Surveillance Frequency Control Program by obtaining a sample of fuel oil based on the applicable ASTM standard, and verifying that total contamination is less than 10 mg/liter when checked based on the applicable ASTM standard.

f. In accordance with the Surveillance Frequency Control Program by:

1. Verify each EDG starts from standby conditions and:

a) In less than or equal to 10 seconds, achieves a voltage greater than 6480 volts (7200 - 720 volts) and a frequency greater than 58.8 Hz (60 - 1.2 Hz).

b) Achieve a steady state voltage greater than 6480 volts but less than 7920 volts and a steady state frequency greater than 58.8 Hz but less than 61.2 Hz.

The EDG shall be started for this test by using one of the following signals:

a) Simulated loss of offsite power by itself.
b) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
c) An ESF actuation test signal by itself.
d) Simulated degraded offsite power by itself.
e) Manual.

2. The generator shall be manually synchronized, loaded to an indicated 4150-4250 kW** in less than or equal to 60 seconds, and operate for at least 60 minutes.

g. In accordance with the Surveillance Frequency Control Program by:

1. Deleted

2. Verifying that on rejection of a load of greater than or equal to 729 kW, the voltage and frequency are maintained at 7200 ± 720 volts and frequency at 60 ± 1.2 Hz.

3. Verifying the generator capability to reject a load of 4250 kW without tripping. The generator voltage shall not exceed 7920 volts during and following the load rejection.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer, and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization of these loads, the steady-state voltage and frequency shall be maintained at 7200 ± 720 volts and 60 ± 1.2 Hz.

h. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 504 rpm in less than or equal to 10 seconds.

i. In accordance with the Surveillance Frequency Control Program, by Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or its equivalent.

j. At least once per 10 years, by performing a pressure test of those portions of the diesel fuel oil system designed to Section III subsection ND of the ASME Code in accordance with Specification 4.0.5.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

a. 125-volt Battery bank No. 1A and its associated full capacity charger.

b. 125-volt Battery bank No. 1B and its associated full capacity charger.

APPLICABILITY: Modes 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN-within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIRMENTS

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

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

b. In accordance with the Surveillance Frequency Control Program and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

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

2. There is no visible corrosion at either terminals or connectors, or the battery connection resistance is less than or equal to the individual connection resistance for the connection types listed below or total battery resistance is less than or equal to 2890 μΩ:

Maximum Individual Battery Connection Resistances		
Connection Type	Number of Connections	Individual Connection Resistance (μΩ)
Inter-cell	56	45
Jumper	3	100
Terminal Plate	2	35

, and

3. The average electrolyte temperature of 10 of the connected cells is ≥ 60°F.

c. In accordance with the Surveillance Frequency Control Program by verifying that:

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

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3. The battery connection resistance is less than or equal to the individual connection resistance for the connection types listed below or total battery resistance is less than or equal to 2890 μΩ:

Maximum Individual Battery Connection Resistances		
Connection Type	Number of Connections	Individual Connection Resistance (μΩ)
Inter-cell	56	45
Jumper	3	100
Terminal Plate	2	35

, and

4. The battery charger will supply at least 300 amperes at 132 volts for at least 8 hours.

d. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. In accordance with the Surveillance Frequency Control Program, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1.d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

ACTION: (Continued)

d. With one D.C. bus not energized from its associated Battery Bank, re-energize the D.C. bus from its associated Battery Bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the busses.

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. Emergency Busses consisting of two 7200 volt and three 480 volt A.C. Emergency Busses.

b. Three 120 volt A.C. Vital Busses energized from their associated inverters connected to their respective D.C. Busses.

c. One 125 volt D.C. Bus energized from its associated battery bank.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, and initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the busses.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 For each containment penetration provided with a penetration conductor overcurrent protective device(s), each device(s) shall be operable.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

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

SURVEILLANCE REQUIREMENTS

4.8.4.1 Protective devices required to be operable as containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE.

a. In accordance with the Surveillance Frequency Control Program:

 1. By verifying that the medium voltage (7.2 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

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

  (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. In accordance with the Surveillance Frequency Control Program by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

ELECTRICAL POWER SYSTEMS

CIRCUIT PROTECTION DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.3 Circuit breakers for non-Class 1E cables located in trays which do not have cable tray covers and which provide protection for cables that if faulted could cause failure in both adjacent, redundant Class 1E cables shall be OPERABLE.

APPLICABILITY: All modes

ACTION:

a. With one or more of the above required non-Class 1E circuit breaker(s) inoperable, within 72 hours, either:

   1. Restore the circuit breaker(s) to OPERABLE status; or
   2. De-energize the circuit breaker(s); or
   3. Establish a one (1) hour roving fire watch for those areas in which redundant systems or components could be damaged.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above required circuit breakers shall be demonstrated OPERABLE.

a. In accordance with the Surveillance Frequency Control Program:

   1. By verifying that the medium voltage (7.2 KV) circuit breakers are OPERABLE by selecting, on a rotating basis) at least 10% of the circuit breakers and performing the following:

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

      (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

      (c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least ten percent (10%) of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breaker's nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturers data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least ten percent (10%) of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. In accordance with the Surveillance Frequency Control Program by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations. |

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2000 ppm.

APPLICABILITY: MODE 6 * with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

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 greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

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 full length control rod 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 system and the refueling canal shall be determined by chemical analysis in accordance with the Surveillance Frequency Control Program.

4.9.1.3 The following valves shall be verified locked closed ** in accordance with the Surveillance Frequency Control Program: 8430, 8454, 8441 and 8439.

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

** Valves may be opened under administrative control to add borated makeup.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6.

ACTION:

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

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the reactor coolant system at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

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

a. A CHANNEL CHECK in accordance with the Surveillance Frequency Control Program,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST in accordance with the Surveillance Frequency Control Program.

REFUELING OPERATIONS

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical a period of time within the acceptable domain of Figure 3.9-1, but not less than 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, immediately suspend all movement of irradiated fuel in the reactor pressure vessel. With the reactor subcritical for greater than 72 hours but not within the acceptable domain of Figure 3.9-1, immediately suspend movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3.1 The reactor shall be determined to have been subcritical for a period of time within the acceptable domain of Figure 3.9-1 by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

4.9.3.2 Prior to moving irradiated fuel from the reactor pressure vessel, and in accordance with the Surveillance Frequency Control Program during movement of irradiated fuel, verify the CCW temperature at the inlet to the Spent Fuel Pool Cooling System heat exchanger is within the acceptable domain of Figure 3.9-1.

3/4.9.5 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.

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.

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.7.1.1 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm in accordance with the Surveillance Frequency Control Program.

4.9.7.1.2 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

* The residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the: vicinity of the reactor pressure vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

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

ACTION:

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

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.7.2.1 At least one residual heat removal loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm in accordance with the Surveillance Frequency Control Program.

4.9.7.2.2 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water in accordance with the Surveillance Frequency Control Program.

* Prior to initial criticality the residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

3/4.9.9 WATER LEVEL - REFUELING CAVITY AND FUEL TRANSFER CANAL

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel or the refueling cavity when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

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

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 control rods.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined in accordance with the Surveillance Frequency Control Program.

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

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODE 1.

ACTION:

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

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

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of the below listed Specifications (a. and b.) shall be performed in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS:

a. Either Specifications 4.2.2.2 or 4.2.2.4 and Specification 4.2.2.5.

b. Specification 4.2.3.2.

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

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

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

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to an ANALOG CHANNEL OPERATION TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specifications 3.4.1.1 may be suspended during the performance of start up and PHYSICS TESTS provided:

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

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set at less than or equal to 25% of RATED THERMAL POWER.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 interlock Setpoint in accordance with the Surveillance Frequency Control Program during start up and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating start up and PHYSICS TESTS.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

With the position indication system inoperable or with more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required rod position indication systems shall be determined to be OPERABLE within 24 hours prior to start of and in accordance with the Surveillance Frequency Control Program thereafter during rod drop time measurements by verifying the demand position indication system and the rod position indication systems agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

*This requirement is not applicable during the initial calibration of rod position indication system provided (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.1 Deleted by Amendment 104.
3.11.1.2 Deleted by Amendment 104 .
3.11.1.3 Deleted by Amendment 104.

3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Condensate Storage Tank
b. Outside Temporary Storage Tank

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1 Deleted by Amendment 104.
4.11.1.2 Deleted by Amendment 104.
4.11.1.3 Deleted by Amendment 104.

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 131,000 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.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 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.

n. <u>Snubber Testing Program</u>

This program conforms to the examination, testing and service life monitoring for dynamic restraints (snubbers) in accordance with 10 CFR 50.55a inservice inspection (ISI) requirements for supports. The program shall be in accordance with the following:

1) This program shall meet 10 CFR 50.55a(g) ISI requirements for supports.

2) The program shall meet the requirements for ISI of supports set forth in subsequent editions of the Code of Record and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that are incorporated by reference in 10 CFR 50.55a(b) subject to limitations and modifications listed in 10 CFR 50.55a(b) and subject to Commission approval.

3) The program shall, as allowed by 10 CFR 50.55a(b)(3)(v), meet Subsection ISTA, "General Requirements," and Subsection ISTD, "Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Nuclear Power Plants," or meet authorized alternatives pursuant to 10 CFR 50.55a(a)(3).

4) The 120-month program updates shall be made in accordance with 10 CFR 50.55a(g)(4), 10 CFR 50.55a(g)(3)(v) and 10 CFR 50.55a(b) (including 10 CFR 50.55a(b)(3)(v)) subject to the limitations and modifications listed therein.

o. <u>Surveillance Frequency Control Program</u>

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.

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

2) 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.

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

ATTACHMENT 5

PROPOSED TS BASES CHANGES

Virgil C. Summer Nuclear Station (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

INSERT 1

in accordance with the Surveillance Frequency Control Program

INSERT 2

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

MOVABLE CONTROL ASSEMBLIES (Continued)

For purposes of determining compliance with Technical Specification 3.1.3.1, any inoperability of full length control rod(s), due to being immovable, invokes ACTION statement "a".

The intent of Technical Specification 3.1.3.1 ACTION Statement "a" is to ensure that before leaving ACTION Statement "a" and utilizing ACTION Statement "c" that the rod urgent failure alarm is illuminated or that an obvious electrical problem is detected in the rod control system by minimal electrical troubleshooting techniques. Expeditious action will be taken to determine if rod immovability is due to an electrical problem in the rod control system.

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

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experience3d during a reactor trip at operating conditions.

Control rod position and OPERABILITY of the rod position indicators are required to be verified ~~on a nominal basis of once per 12 hours~~ with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

INSERT 1

POWER DISTRIBUTION LIMIT

BASES

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty to the measured $F_{\Delta H}^{N}$ value.

INSERT 2

~~The 12 hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation specified on the RCS Total Flow Rate Versus R Figure in the COLR.~~

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt power ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors or the PDMS are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum of DNBR in the core at or above the design limit throughout each analyzed transient. The maximum indicated T_{avg} limit of 589.2°F and the minimum indicated pressure limit of 2206 psig correspond to analytical limits of 591.4°F and 2185 psig respectively, read from control board indications.

INSERT 2

~~The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.~~

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Protection System and Engineered Safety Feature Actuation System Instrumentation and interlocks ensure that 1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoints, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functions capability is available from diverse parameters. Insert 1

Insert 2

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. ~~Specified surveillance intervals have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for Reactor Protection Instrumentation System," and supplements to that report.~~ ~~Specified surveillance and maintenance~~ outage times have been determined in accordance with WCAP-14333-P-A, Rev. 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," and Westinghouse letter CGE-05-46. ~~Specified surveillance intervals and~~ RTB outage times have been determined in accordance with WCAP-15376-P-A, Rev. 1, "Risk-Informed Assessment of the RTS and ESFAS ~~Surveillance~~ Test Intervals and Reactor Trip Breaker Test and Completion Times," dated March 2003. ~~Surveillance intervals and~~ out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. ~~The Slave Relay Test is performed on an 18 month frequency that is specific to Westinghouse AR relays. This test frequency is based on relay reliability assessments presented in WCAP-13877-P-A, "Reliability Assessment of Westinghouse Type AR Relays Used as SSPS Slave Relays," that is dependent on the qualified life and environmental conditions of the AR relays. Replacement relays other than Westinghouse type AR or reconciled Cutler-Hammer relays will require further analysis and NRC approval.~~

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Consistent with the requirement in Regulatory Guide 1.177 to include Tier 2 insights into the decision-making process before taking equipment out of service, restrictions on concurrent removal of certain equipment when a logic train is inoperable for maintenance are included (note that these restrictions do not apply when a logic train is being tested under the 4-hour bypass Note). Entry into Actions 12, 14, 21, or 25 is not a typical, pre-planned evolution during power operation, other than for surveillance testing. Since Actions 12, 14, 21, or 25 are typically entered due to equipment failure, it follows that some of the following restrictions may not be met at the time of entry into Actions 12, 14, 21, or 25. If this situation were to occur during the 24-hour AOT of Actions 12, 14, 21, or 25, the configuration risk assessment procedure will assess the emergent condition and direct activities to restore the inoperable logic train and exit Actions 12, 14, 21, or 25, or fully implement these restrictions, or perform a unit shutdown, as appropriate from a risk management perspective. The following restrictions will be observed:

- To preserve ATWS mitigation capability, activities that degrade the availability of the emergency feedwater system, RCS pressure relief system (pressurizer PORVs and safety valves), AMSAC, or turbine trip should not be scheduled when a logic train is inoperable for maintenance.

INSTRUMENTATION

BASES

3/4.3.3.9 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.11 POWER DISTRIBUTION MONITORING SYSTEM (PDMS)

The Power Distribution Monitoring System (PDMS) provides core monitoring of the limiting parameters. The PDMS continuous core power distribution measurement methodology begins with the periodic generation of a highly accurate 3-D nodal simulation of the current reactor power distribution. The simulated reactor power distribution is then continuously adjusted by nodal and thermocouple calibration factors derived from an incore power distribution measurement obtained using the incore movable detectors to produce a highly accurate power distribution measurement. The nodal calibration factors are updated ~~at least once every 180 Effective Full Power Days (EFPD).~~ INSERT 1 Between calibrations, the fidelity of the measured power distribution is maintained via adjustment to the calibrated power distribution provided by continuously input plant and core condition information. The plant and core condition data utilized by the PDMS is cross checked using redundant information to provide a robust basis for continued operation. The loop inlet temperature is generated by averaging the respective temperatures from each of the loops, excluding any bad data. The core exit thermocouples provide many readings across the core and by the nature of their usage with the PDMS, smoothing of the measured data and elimination of bad data is performed with the Surface Spline fit. PDMS uses the NIS Power Range excore detectors to provide information on the axial power distribution. Hence, the PDMS averages the data from the four Power Range excore detectors and eliminates any bad excore detector data.

The bases for the operability requirements of the PDMS is to provide assurance of the accuracy and reliability of the core parameters measured and calculated by the PDMS core power distribution monitor function. These requirements fall under four categories:

1. Assure an adequate number of operable critical sensors.
2. Assure sufficiently accurate calibration of these sensors.
3. Assure an adequate calibration data base regarding the number of data sets.
4. Assure the overall accuracy of the calibration.

INSTRUMENTATION

BASES

POWER DISTRIBUTION MONITORING SYSTEM (PDMS) (Continued)

The minimum number of required plant and core condition inputs includes the following:

1. Control Bank Positions.

2. At least 50% of the cold leg temperatures.

3. At least 75% of the signals from the Power Range excore detector channels (comprised of a top and bottom detector section).

4. Reactor Power Level.

5. A minimum number and distribution of operable core exit thermocouples.

6. A minimum number and distribution of measured fuel assembly power distribution information obtained using the incore movable detectors is incorporated in the nodal model calibration information.

The sensor calibration of items 1., 2., 3., and 4. above are covered under other specifications. Calibration of the core exit thermocouples is accomplished in two parts. The first being a sensor specific correction to K-type thermocouple temperature indications based on data from a cross calibration of the thermocouple temperature indications to the average RCS temperature measured via the RTDs under isothermal RCS conditions. The second part of the thermocouple calibration is the generation of thermocouple flow mixing factors which cause the radial power distribution measured via the thermocouples to agree with the radial power distribution from a full core flux map measured using the incore movable detectors. This calibration is updated ~~at least once every 180 EFPD.~~

INSERT 1

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION (Continued)

RHR system piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the RHR loops and may also prevent water hammer, pump cavitation, and pumping of non-condensable gas into the reactor vessel.

Selection of RHR System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrument drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walkdowns to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as standby versus operating conditions.

The RHR System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the RHR System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

RHR System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, plant configuration, or personnel safety. For these locations, alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

Surveillance Requirement 4.4.1.3.4 is modified by a Note that states the Surveillance Requirement is not required to be performed until 12 hours after entering MODE 4. In a rapid shutdown, there may be insufficient time to verify all susceptible locations prior to entering MODE 4.

Insert 2

~~The 31 day frequency for ensuring locations are sufficiently filled with water takes into consideration the gradual nature of gas accumulation in the RHR System piping and the procedural controls governing system operation.~~

REACTOR COOLANT SYSTEM

BASES

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operating relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will be performed in accordance with the provisions of the ASME OM Code.

INSERT 2

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady sate envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. ~~The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation.~~ The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVE (PORVs)

The pressurizer power operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. The PORVs and block valves may be used to depressurize the RCS when normal pressurizer spray is unavailable. Operation of the air operated PORVs minimizes the undesirable opening of the spring loaded pressurizer code safety valves. Each PORV has a remotely controlled motor-operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The series arrangement of the PORV and its associated block valve permit surveillance while at power.

REACTOR COOLANT SYSTEM

BASES

OPERATIONAL LEAKAGE (Continued)

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity and containment sump level. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Reactor Coolant System, Leakage Detection Systems."

Part (d) notes that this SR is not applicable to primary-to-secondary leakage because leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

~~The 72 hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.~~ Insert 2

4.4.6.2.2 This Surveillance Requirement verifies RCS Pressure Isolation Valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

~~Pressure Isolation Valve (PIV) testing is being conducted on a performance based testing interval. Performance-based is defined as a PIV which demonstrates good performance for two consecutive outages and may have its test interval extended to every third refueling outage (to a maximum of 60 months). A PIV, which has demonstrated leakage less than its TS allowable leakage limit for two consecutive outages is classified as a good performer. In the event of a PIV leakage test failure, PIV testing would require the component to return to the initial interval of every refueling outage, or two years until good performance is re-established.~~ Insert 1

Those PIVs listed in TS Table 3.4-1 will be demonstrated OPERABLE by verifying their leakage rates are within TS allowable leakage limits via testing performed ~~at a frequency based on their "good" performance, ranging from every refueling outage to every third refueling outage (to a maximum of 60 months).~~

4.4.6.2.3 This Surveillance Requirement verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one steam generator. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5 should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 2. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator.

The Surveillance Requirement is modified by a note, which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For Reactor Coolant System primary-to-secondary leakage determination, steady state is defined as stable Reactor Coolant System pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows. Insert 2

~~The frequency of 72 hours is a reasonable interval to trend primary to secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents.~~ The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Reference 2).

References

1. NEI 97-06, "Steam Generator Program Guidelines"

2. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines"

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits. INSERT 2

~~The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.~~

REACTOR COOLANT SYSTEM

BASES

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Virgil C. Summer site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

INSERT 2

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. ~~The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action.~~ Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cool down are limited by curves developed using the methodology from Westinghouse Topical Report, WCAP-14040-NP-A, updated to include the requirements of the 1998 ASME Boiler and Pressure Vessel Code, Section XI, through the 2000 Addenda, Appendix G, along with ASME Code Case N-641.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3.

    a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

REACTOR COOLANT SYSTEM

BASES

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two RHRSRVs or an RCS vent opening of at least 2.7 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 300°F. Either RHRSRV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of an HPSI pump and its injection into a water solid RCS.

The limitation for a maximum of one charging pump to be capable of injecting into the RCS, and the Surveillance Requirement to verify at least two charging pumps are demonstrated to be INOPERABLE ~~at least once per 31 days,~~ while the RCS is below 300°F, provides assurance that a mass addition transient can be mitigated by a single RHR suction relief valve.

INSERT 1

ECCS SUBSYSTEMS (Continued)

(e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

Surveillance Requirement 4.5.2.b.1) is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing a dedicated individual at the system vent path who is in continuous communication with the operators in the control room. The individual will have a method to rapidly close the system vent flow path if directed.

Insert 2

~~The 31 day frequency for Surveillance Requirement 4.5.2.b.2) taken into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.~~

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of either a LOCA, a steamline break or inadvertent RCS depressurization. The limits of RWST minimum volume and boron concentration ensure 1) that sufficient water is available within containment to permit recirculation cooling flow to the core, 2) that the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS, Spray Additive Tank (SAT), containment spray system piping and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1), 3) that the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area ≥ 3.0 sq. ft.) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump post-LOCA with all control rods assumed to be out (ARO), 4) long term subcriticality following a steamline break assuming ARI-1 and preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (Post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

REACTOR BUILDING SPRAY SYSTEM (Continued)

Containment Spray System flow path piping and components have the potential to develop voids and pockets of entrained gases. Preventing and managing gas intrusion and accumulation is necessary for proper operation of the containment spray trains and may also prevent a water hammer and pump cavitation.

Selection of Containment Spray System locations susceptible to gas accumulation is based on a review of system design information, including piping and instrument drawings, isometric drawings, plan and elevation drawings, and calculations. The design review is supplemented by system walkdowns to validate the system high points and to confirm the location and orientation of important components that can become sources of gas or could otherwise cause gas to be trapped or difficult to remove during system maintenance or restoration. Susceptible locations depend on plant and system configuration, such as standby versus operating conditions.

The Containment Spray System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the Containment Spray System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

Containment Spray System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, plant configuration, or personnel safety. For these locations, alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval.

Insert 2

~~The 31 day frequency for Surveillance Requirement 4.6.2.1.a.1) takes into consideration the gradual nature of gas accumulation in the Containment Spray System piping and the procedural controls governing system operation.~~

Surveillance Requirement 4.6.2.1.a.1) is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing a dedicated individual at the system vent path who is in continuous communication with the operators in the control room. The individual will have a method to rapidly close the system vent flow path if directed.

PLANT SYSTEMS

BASES

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

SR 4.7.6.b

Insert 1

Standby systems should be checked periodically to ensure that they function properly. ~~As the environment and normal operating conditions on this system are not too severe, testing each train once every month~~ provides an adequate check of this system. The VCSNS CREFS does not have heaters and each train need only be operated for a minimum of 15 minutes to demonstrate the function of the system. ~~The 31 day Frequency is based on the reliability of the equipment and the two train redundancy.~~

INSERT 2

SR 4.7.6.c

This SR verifies that the required CREFS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP implementing procedures.

SR 4.7.6.d

This SR verifies that each CREFS train starts and operates on an actual or simulated actuation signal. ~~The Frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle.~~

INSERT 2

SR 4.7.6.e

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem whole body or its equivalent to any part of the body and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, ACTION 3.7.6.a.2 must be entered. Action 3.7.6.a.2 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 5)

BASES

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

Insert 1

The surveillance requirements applicable to the medium voltage (7.2 kV) circuit breakers provide assurance of breaker reliability by testing 10% of the circuit breakers and their associated relays and control circuits ~~at least once per 18 months~~ on a rotating basis. The only medium voltage conductors that penetrate the containment are for the three RCP motors. The breakers associated with these conductors are listed on FSAR Figure 8G-2. A failure of any portion of the integrated system (relays, control circuit and circuit breaker) results in an inoperable circuit breaker. A retest in accordance with Surveillance Requirement 4.8.4.1.a.1.(c) of an additional representative sample of at least 10% of all circuit breakers equates to a retest of one circuit breaker and its associated relays and control circuit. Retests are performed until no more failures are found or all breakers have best tested.

The surveillance requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The surveillance requirements of the circuit breakers for non-Class 1E cables located in trays which do not have cable tray covers and which provide protection for cables that, if faulted, could cause failure in both adjacent, redundant Class 1E cables ensures that the integrity of Class 1E cables is not compromised by the failure of protection devices to operate in the non-Class 1E cables.

RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION (Continued)

The RHR System is OPERABLE when it is sufficiently filled with water. Acceptance criteria are established for the volume of accumulated gas at susceptible locations. If accumulated gas is discovered that exceeds the acceptance criteria for the susceptible location (or the volume of accumulated gas at one or more susceptible locations exceeds an acceptance criteria for gas volume at the suction or discharge of a pump), the Surveillance is not met. If it is determined by subsequent evaluation that the Containment Spray System is not rendered inoperable by the accumulated gas (i.e., the system is sufficiently filled with water), the Surveillance may be declared met. Accumulated gas should be eliminated or brought within the acceptance criteria limits.

RHR System locations susceptible to gas accumulation are monitored and, if gas is found, the gas volume is compared to the acceptance criteria for the location. Susceptible locations in the same system flow path which are subject to the same gas intrusion mechanisms may be verified by monitoring a representative subset of susceptible locations. Monitoring may not be practical for locations that are inaccessible due to radiological or environmental conditions, plant configuration, or personnel safety. For these locations, alternative methods (e.g., operating parameters, remote monitoring) may be used to monitor the susceptible location. Monitoring is not required for susceptible locations where the maximum potential accumulated gas void volume has been evaluated and determined to not challenge system OPERABILITY. The accuracy of the method used for monitoring the susceptible locations and trending of the results should be sufficient to assure system OPERABILITY during the Surveillance interval. Insert 2

~~The 31 day frequency for ensuring locations are sufficiently filled with water takes into consideration the gradual nature of gas accumulation in the RHR System piping and the procedural controls governing system operation.~~

3/4.9.8 DELETED BY AMENDMENT 183

3/4.9.9 WATER LEVEL - REACTOR VESSEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99.5% of the assumed 16% I-131 and 10% other halogens gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

ATTACHMENT 6

PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION

Virgil C. Summer Nuclear Station (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION

Description of Amendment Request:

This amendment request involves the adoption of an approved change to the standard Technical Specifications (STS) for Westinghouse Plants (NUREG-1431) [8.30], to allow relocation of specific TS surveillance frequencies to a licensee-controlled program. The proposed change is described in Technical Specification Task Force (TSTF) Traveler TSTF-425, Revision 3 (Rev. 3) (ADAMS Accession No. ML090850642) [8.3], "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b" and was described in the Notice of Availability published in the Federal Register on July 6, 2009 [8.4].

The proposed changes are consistent with NRC-approved Industry/Technical Specification Task Force (TSTF) Traveler, TSTF-425, Rev. 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b." [8.3]. The proposed change relocates surveillance frequencies to a licensee-controlled program, the SFCP. This change is applicable to licensees using probabilistic risk guidelines contained in NRC-approved NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," (ADAMS Accession No. ML071360456) [8.5].

Basis for proposed no significant hazards consideration:

As required by 10 CFR 50.91(a) [8.8], the DESC analysis of the issue of no significant hazards consideration is presented below:

Dominion Energy South Carolina (DESC) has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," [8.31] as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

The proposed change relocates the specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the Technical Specifications for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any

mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

2. Does the proposed change create the possibility of a new or different kind of accident from any accident 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 plan (*i.e.*, no new 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 or malfunction 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, DESC will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Rev.1 [8.5] 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 RG 1.177 [8.32].

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above, DESC concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), Issuance of Amendment [8.31].

ATTACHMENT 7

PROPOSED VCSNS TS CHANGES VERSES TSTF-425 CROSS-REFERENCE

Virgil C. Summer Nuclear Station (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.1.1.1.1.b	N/A	VCSNS TS specific periodic SR for verification of shutdown margin by rod control bank position.	12 Hours	N/A
4.1.1.1.2	3.1.2.1	Core Reactivity/Verify core reactivity	31 Effective Full Power Days (EFPD)	Yes
4.1.1.2.b	3.1.1.1	Core Reactivity/Verify shutdown margin	24 Hours	Yes
4.1.2.1.1	N/A	VCSNS TS specific periodic SR for verification of valve positions in boration flow paths.	31 Days	N/A
4.1.2.2.a	N/A	VCSNS TS specific periodic SR for verification of valve positions in boration flow paths.	31 Days	N/A
4.1.2.2.b	N/A	VCSNS TS specific periodic SR for verification of flow rate from boration flow paths.	18 Months	N/A
4.1.2.5.a	N/A	VCSNS TS specific periodic SR for verification of borated water sources volume, boron concentration and temperature.	7 Days	N/A
4.1.2.5.b	N/A	VCSNS TS specific periodic SR for refueling water storage tank (RWST) temperature.	24 Hours	N/A
4.1.2.6.a	N/A	VCSNS TS specific periodic SR for verification of borated water sources volume, boron concentration and temperature.	7 Days	N/A
4.1.2.6.b	N/A	VCSNS TS specific periodic SR for RWST temperature.	24 Hours	N/A
4.1.3.1.1	3.1.4.1	Rod Group Alignment Limits/Verify individual rod positions	12 Hours	Yes
4.1.3.1.2	3.1.4.2	Rod Group Alignment Limits	31 Days	Yes
4.1.3.2	N/A	VCSNS TS specific periodic SR for verification of rod position indication compared to demand position.	12 Hours	N/A

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.1.3.3	N/A	VCSNS specific periodic SR for ANALOG CHANNEL OPERATIONAL TEST of rod position indicators.	18 Months	N/A
4.1.3.4.c	N/A	VCSNS specific SR for verification of rod drop times.	18 Months	N/A
4.1.3.5.b	3.1.5.1	Shutdown Bank Insertion Limits/Verify each shutdown bank within limits	12 Hours	Yes
4.1.3.6	3.1.6.2	Control Bank Insertion Limits/Verify each control bank within limits	12 Hours	Yes
4.2.1.1.a	3.2.3.1	AFD/Verify AFD for each OPERABLE ex-core detector	7 Days	Yes
4.2.1.3	3.2.3.3	AFD/Determine target flux difference	92 EFPD	Yes (only for 3.2.3A.3)
4.2.1.4	3.2.3.2	AFD/Update target flux difference	31 EFPD	Yes(only for 3.2.3A.2)
4.2.2.2.d.2	3.2.1.1	$F_Q(Z)$/Verify $F_Q^C(Z)$ in limit (VCSNS TS has separate SRs for Relaxed Axial Offset Control (RAOC) and base load operations)	31 EFPD	Yes
4.2.2.4.d.2	3.2.1.1	$F_Q(Z)$/Verify $F_Q^C(Z)$ in limit (VCSNS TS has separate SRs for Relaxed Axial Offset Control (RAOC) and base load operations)	31 EFPD	Yes
4.2.3.2.b	3.2.2.1	$F_{\Delta H}^N(Z)$/Verify $F_{\Delta Q}^N(Z)$ in limit	31 EFPD	Yes
4.2.3.3	3.4.1.3	RCS Pressure, Temperature, and Flow DNB Limits/Verify RCS total flowrate	12 Hours	Yes
4.2.3.4	N/A	VCSNS TS specific periodic SR for RCS flow rate indicators channel calibration.	18 Months	N/A
4.2.3.5	3.4.1.4	RCS Pressure, Temperature, and Flow DNB Limits/Verify RCS total flowrate by precision heat balance	18 Months	Yes
4.2.4.1.a	3.2.4.1	QPTR/Verify QPTR in limit by calculation	7 Days	Yes

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.2.4.1.b	3.2.4.1	QPTR/Verify QPTR in limit by calculation	12 Hours	Yes
4.2.4.2	3.2.4.2	QPTR/Verify QPTR in limit using movable injection detectors	12 Hours	Yes
4.2.5	3.4.1.1 and 3.4.1.2	RCS Pressure, Temperature, and Flow DNB Limits/Verify pressurizer pressure (3.4.1.1) and /Verify RCS average temperature (3.4.1.2)	12 Hours	Yes
4.3.1.1/Table 4.3-1	3.3.1.1 – 3.3.1.15	Reactor Trip System (RTS) Instrumentation/Various instrumentation checks, operational tests, calibration	Various	Yes
4.3.1.2	3.3.1.16	Reactor Trip System (RTS) Instrumentation/Various instrumentation response time verifications	18 Months	Yes
4.3.2.1/Table 4.3-2	3.3.2.1 – 3.3.2.9	Engineered Safety Feature Actuation System (ESFAS) Instrumentation/Various instrumentation checks, operational tests, calibration	Various	Yes
4.3.2.2	3.3.2.10	Engineered Safety Feature Actuation System (ESFAS) Instrumentation/ESFAS instrumentation response time verifications	18 Months	Yes
4.3.3.1/Table 4.3-3	3.3.7.1- 3.3.7.9	Control Room Emergency Filtration System (CREFS) Actuation Instrumentation/Various instrumentation checks, operational tests, calibration	Various	Yes
4.3.3.1/Table 4.3-3	3.3.8.1 – 3.3.8.5	FBACS Actuation Instrumentation/Various instrumentation checks, operational tests, calibration	Various	Yes
4.3.3.1	N/A	VCSNS TS specific periodic SRs for testing of radiation monitoring instrumentation channels other than CREFS Actuation.	Various	N/A

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.3.3.2	N/A	VCSNS TS specific periodic SR for testing of the movable incore detectors.	24 Hours	N/A
4.3.3.4/Table 4.3-5	N/A	VCSNS TS specific periodic SR for testing of meteorological instrumentation.	Various	N/A
4.3.3.5/Table 4.3-6	3.3.4.1, 3.3.4.3	Remote Shutdown System/CHANNEL CHECK (3.3.4.1) and CHANNEL CALIBRATION (3.3.4.3)	Various	Yes
4.3.3.6	3.3.3.1, 3.3.3.2	Post Accident Monitoring (PAM) Instrumentation/CHANNEL CHECK (3.3.3.1) and CHANNEL CALIBRATION (3.3.3.2)	1 month and every refueling outage	Yes
4.3.3.9/Table 4.3-9	N/A	VCSNS TS specific periodic SRs for testing of explosive gas monitoring instrumentation.	Various	N/A
4.3.3.10.a	N/A	VCSNS TS specific periodic SR for testing of loose-part detection instrumentation.	24 Hours	N/A
4.3.3.10.b	N/A	VCSNS TS specific periodic SR for testing of loose-part detection instrumentation.	31 Days	N/A
4.3.3.10.c	N/A	VCSNS TS specific periodic SR for testing of loose-part detection instrumentation.	18 Months	N/A
4.3.3.11.2.a	N/A	VCSNS TS specific periodic SR for testing of the power distribution monitoring system.	180 EFPD	N/A
4.3.3.11.2.b	N/A	VCSNS TS specific periodic SR for testing of the power distribution monitoring system.	31 EFPD	N/A
4.4.1.1	3.4.4.1	RCS Loops – MODES 1 and 2/Verify each loop in operation	12 Hours	Yes
4.4.1.2.1	3.4.5.3	RCS Loops – MODE 3/Verify breaker alignment and indicated power	7 Days	Yes
4.4.1.2.2	3.4.5.1	RCS Loops – MODE 3/Verify required loops in operation	12 Hours	Yes

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.4.1.2.3	3.4.5.2	RCS Loops – MODE 3/Verify steam generator water levels	12 Hours	Yes
4.4.1.3.1	3.4.6.3	RCS Loops – MODE 4/Verify breaker alignment and indicated power	7 Days	Yes
4.4.1.3.2	3.4.6.2	RCS Loops – MODE 4/Verify steam generator water levels	12 Hours	Yes
4.4.1.3.3	3.4.6.1	RCS Loops – MODE 4/Verify required loop in operation	12 Hours	Yes
4.4.1.3.4	N/A	VCSNS TS specific SR to verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	31 Days	N/A
4.4.1.4.1.1	3.4.7.2	RCS Loops – MODE 5, Loops Filled/Verify steam generator water levels	12 Hours	Yes
4.4.1.4.1.2	3.4.7.1	RCS Loops – MODE 5, Loops Filled/Verify required RHR loop in operation	12 Hours	Yes
4.4.1.4.1.3	N/A	VCSNS TS specific SR to verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	31 Days	N/A
4.4.1.4.2.1	3.4.8.1	RCS Loops – MODE 5, Loops Not Filled/Verify required RHR loop in operation	12 Hours	Yes
4.4.1.4.2.2	N/A	VCSNS TS specific SR to verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	31 Days	N/A
4.4.3.1	3.4.9.1	Pressurizer/Verify pressurizer water level	12 Hours	Yes
4.4.3.2	3.4.9.2	Pressurizer/Verify pressurizer heater capacity	92 Days	Yes
4.4.4.1	3.4.11.2	Pressurizer PORVs/Cycle each PORV	18 Months	Yes
4.4.4.2	3.4.11.1	Pressurizer PORVs/Cycle each block valve	92 Days	Yes

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.4.6.1.b	3.4.15.3	RCS Leakage Detection Instrumentation/CHANNEL CALIBRATION of containment sump monitor	18 Months	Yes
4.4.6.1.d	3.4.15.5	RCS Leakage Detection Instrumentation/CHANNEL CALIBRATION of containment air cooler condensate flow rate monitor	18 Months	Yes
4.4.6.2.1.a	N/A	VCSNS TS specific periodic SR for monitoring reactor building atmosphere radioactivity.	12 Hours	N/A
4.4.6.2.1.b	N/A	VCSNS TS specific periodic SR for monitoring reactor building sump inventory.	12 Hours	N/A
4.4.6.2.1.c	3.5.5.1	Verify manual valve adjustment (VCSNS SR is in RCS Leakage section).	31 Days	Yes
4.4.6.2.1.d	3.4.13.1	Perform water inventory balance	72 Hours	Yes
4.4.6.2.1.e	N/A	VCSNS TS specific periodic SR for monitoring reactor head flange leakoff temperature	24 Hours	N/A
4.4.6.2.2.a	3.4.14.1	RCS PIV Leakage/Verify leakage from each PIV in limit	Max 60 Months	Yes
4.4.6.2.3	3.4.13.2	RCS Operational Leakage/Verify primary to secondary LEAKAGE in limit	72 Hours	Yes
4.4.7/Table 4.4-3	N/A	VCSNS TS specific periodic SR for verification of RCS chemistry parameters per Table 4.4-3.	Various	N/A
4.4.8/Table 4.4-4	3.4.16.1- 3.4.16.3	RCS Specific Activity/Verify RCS gross specific activity	Various	Yes
4.4.9.1.1	3.4.3.1	RCS P/T Limits/Verify RCS pressure, temperature, and heat-up and cooldown rates	30 Minutes	Yes
4.4.9.2	N/A	VCSNS TS specific periodic SRs to verify pressurizer temperatures and spray water temperature differential.	30 Minutes and 12 Hours	N/A

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.4.9.3.1.a	3.4.12.4	LTOP System/Verify required RHR suction valves open	72 Hours	Yes
4.4.9.3.1.c	N/A	VCSNS TS specific periodic SR for RHR relief valve setpoint verification.	18 Months	N/A
4.4.9.3.2	3.4.12.5	LTOP System/Verify RCS vent open	12 Hours	Yes
4.4.9.3.3	3.4.12.1	LTOP System/Verify a maximum number of HPI pumps capable of injecting into RCS	31 Days	Yes
4.5.1.1.a.1	3.5.1.2	Accumulators/Verify borated water volume	12 Hours	Yes
4.5.1.1.a.1	3.5.1.3	Accumulators/Verify nitrogen cover pressure	12 Hours	Yes
4.5.1.1.a.2	3.5.1.1	Accumulators/Verify each isolation valve open	12 Hours	Yes
4.5.1.1.b	3.5.1.4	Accumulators/Verify boron concentration	31 Days	Yes
4.5.1.1.c	3.5.1.5	Accumulators/Verify isolation valve power removed	31 Days	Yes
4.5.1.1.d	N/A	VCSNS TS specific periodic SR for verification of accumulator isolation valve automatic opening.	18 Months	N/A
4.5.2.a	3.5.2.1	ECCS – Operating/Verify valve position and power to operator removed	12 Hours	Yes
4.5.2.b.1	3.5.2.2	ECCS – Operating/Verify valve positions	31 Days	Yes
4.5.2.b.2	3.5.2.3	ECCS – Operating/Verify piping filled with water	31 Days	Yes
4.5.2.d.1	3.4.14.2	RCS PIV Leakage/Verify RHR System autoclosure interlock prevents valves from opening	18 Months	Yes
4.5.2.d.2	3.5.2.8	ECCS – Operating/Verify containment sump free of debris, structural distress or abnormal corrosion	18 Months	Yes
4.5.2.e.1	3.5.2.5	ECCS – Operating/Verify automatic valve actuation	18 Months	Yes

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.5.2.e.2	3.5.2.6	ECCS – Operating/Verify pump start on actuation signal	18 Months	Yes
4.5.2.g.2	3.5.2.7	ECCS – Operating/Verify throttle valve position	18 Months	Yes
4.5.3.2	3.4.12.1	LTOP System/Verify a maximum number of HPI pumps capable of injecting into RCS	31 Days	Yes
4.5.4.a.1	3.5.4.2	RWST/Verify RWST volume	7 Days	Yes
4.5.4.a.2	3.5.4.3	RWST/Verify RWST boron concentration	7 Days	Yes
4.5.4.b	3.5.4.1	RWST/Verify RWST temperature	24 Hours	Yes
4.6.1.1.a	3.6.3.3	Containment Isolation Valves/Verify outside containment isolation valves	31 Days	Yes
4.6.1.3.c	3.6.2.2	Containment Air Locks/Verify only one door can be opened at a time	6 Months	Yes
4.6.1.4	3.6.4.1	Containment Pressure/Verify containment pressure in limit	12 Hours	Yes
4.6.1.5	3.6.5A.1	Containment Temperature/Verify containment temperature in limit	24 Hours	Yes
4.6.1.7.1.a	N/A	VCSNS TS specific periodic SR for 36-inch purge valve closure verification.	24 Hours	N/A
4.6.1.7.1.b	3.6.3.1	Containment Isolation Valves/Verify large purge valves sealed closed	31 Days	Yes
4.6.1.7.2	3.6.3.2	Containment Isolation Valves/Verify small purge valves closed when required	7 Days	Yes
4.6.1.7.3	3.6.3.7	Containment Isolation Valves/Leakage rate testing for purge valves	30 Months	Yes
4.6.2.1.a.1	3.6.6.1	Containment Spray and Cooling Systems/Verify valve positions	31 Days	Yes

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.6.2.1.a.2	N/A	VCSNS TS specific SR for verification of Containment Spray locations susceptible to gas accumulation are sufficiently filled with water.	31 Days	N/A
4.6.2.1.c.1	3.6.6.5	Containment Spray and Cooling Systems/Verify actuation of automatic valves	18 Months	Yes
4.6.2.1.c.2	3.6.6.6	Containment Spray and Cooling Systems/Verify pump start on actuation signal	18 Months	Yes
4.6.2.2.a	3.6.7.1	Spray Additive System/Verify valve positions	31 Days	Yes
4.6.2.2.b.1	3.6.7.2	Spray Additive System/Verify tank volume	6 Months	Yes
4.6.2.2.b.2	3.6.7.3	Spray Additive System/Verify solution concentration	6 Months	Yes
4.6.2.2.c	3.6.7.4	Spray Additive System/Verify automatic valve actuation	18 Months	Yes
4.6.2.2.d	3.6.7.5	Spray Additive System/Verify flow rate	5 years	Yes
4.6.2.3.a.1	3.6.6.2	Containment Spray and Cooling Systems/Operate each containment cooling train 15 minutes	31 Days	Yes
4.6.2.3.b.1	3.6.6.7	Containment Spray and Cooling Systems/Verify cooling train start on actuation signal	18 Months	Yes
4.6.2.3.b.2	3.6.63	Containment Spray and Cooling Systems/Verify cooling water flowrate	18 Months	Yes
4.6.2.3.b.3	N/A	VCSNS TS specific periodic SR for verification of containment cooling automatic valve operation.	18 Months	N/A
4.6.2.3.b.4	N/A	VCSNS TS specific periodic SR for verification of service water system booster pump operation.	18 Months	N/A
4.6.3.a	3.6.11.1	Iodine Cleanup System/Operate each ICS train	31 Days	Yes

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference

VCSNS TS [8.1] SR	Corresponding SR in NUREG-1431 [8.30]	TSTF-425 Section Title/SR Description	VCSNS TS [8.1] Surveillance Frequency	SR Frequency modified by TSTF-425 [8.3]
4.6.3.c.1	3.6.11.4	Iodine Cleanup System/Verify ICS bypass filter damper can be operated	18 Months	Yes
4.6.3.c.2	3.6.11.3	Iodine Cleanup System/Verify ICS actuation	18 Months	Yes
4.6.4.2	3.6.3.8	Containment Isolation Valves/Verify actuation of automatic isolation valves	18 Months	Yes
4.7.1.2.a.3 and 4.7.1.2.a.4	3.7.5.1	AFW System/Verify valve position	31 Days	Yes
4.7.1.2.a.5	N/A	VCSNS TS specific periodic SR for locked valve position verification.	31 Days	N/A
4.7.1.2.b	N/A	VCSNS TS specific periodic SR for instrument air supply line check valve closure on loss of normal instrument air supply verification.	3 Months	N/A
4.7.1.2.c.1	3.7.5.4	AFW System/Verify automatic pump start	18 Months	Yes
4.7.1.2.c.2	N/A	VCSNS TS specific periodic SR for verification of capability to hold close emergency feedwater control valves using accumulators when normal instrument air supply is unavailable.	18 Months	N/A
4.7.1.2.c.3	N/A	VCSNS TS specific periodic SR for verification of capability to manually close steam supply valve to turbine driven emergency feedwater pump using accumulators when normal instrument air supply is unavailable.	18 Months	N/A
4.7.1.2.c.4	3.7.5.3	AFW System/Verify automatic valve actuation	18 Months	Yes
4.7.1.3.1	3.7.6.1	Condensate Storage Tank (CST)/Verify CST level	12 Hours	Yes
4.7.1.3.2	N/A	VCSNS TS specific periodic SR for service water system pressure verification.	12 Hours	N/A

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.7.1.4/Table 4.7-1, item 1	N/A	VCSNS TS specific periodic SR in Table 4.7-1 item 1 for gross activity determination.	72 Hours	N/A
4.7.1.4/Table 4.7-1, item 2 which has two SRs are based on gross activity levels	3.7.18.1	Secondary Specific Activity/verify DOSE EQUIVALENT I-131	Various	Yes
4.7.2	N/A	VCSNS TS specific periodic SR for steam generator pressure verification.	1 Hour	N/A
4.7.3.a	3.7.7.1	Component Cooling Water (CCW) System/Verify valve position	31 Days	Yes
4.7.4.a	3.7.8.1	Service Water System (SWS)/Verify valve positions	31 Days	Yes
4.7.5	3.7.9.1 and 3.7.9.2	Ultimate Heat Sink (UHS)/Verify UHS water level (3.7.9.1) and Verify UHS water temperature	24 Hours	Yes
4.7.6.a	N/A	VCSNS TS specific periodic SR for CREFS control room air temperature verification.	12 Hours	N/A
4.7.6.b	3.7.10.1	Control Room Emergency Filtration System (CREFS)/Operate each CREFS train	31 Days	Yes
4.7.6.d	3.7.10.3	Control Room Emergency Filtration System (CREFS)/Verify automatic actuation of CREFS	18 Months	Yes
4.7.8.2.a	N/A	VCSNS TS specific periodic SR for sealed sources leakage testing.	6 Month	N/A
4.7.9	N/A	VCSNS TS specific periodic SR for area temperature checks.	12 Hours	N/A
4.7.10	3.7.15.1	Fuel Storage Pool Water Level/Verify pool water level	7 Days	Yes
4.7.13	3.7.16.1	Fuel Storage Pool Boron Concentration/Verify pool boron concentration	72 Hours	Yes

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.8.1.1.1	3.8.1.1	AC Sources – Operating/Offsite circuit breaker alignment and power available	7 Days	Yes
4.8.1.1.2.a.1	3.8.1.4	AC Sources – Operating/Verify day tank level	31 Days	Yes
4.8.1.1.2.a.2	3.8.1.6	AC Sources – Operating/Verify fuel oil transfer system operation	31 Days	Yes
4.8.1.1.2.a.3	3.8.1.2	AC Sources – Operating/DG start and achieve voltage and power available	31 Days	Yes
4.8.1.1.2.a.4	3.8.1.3	AC Sources – Operating/DG synchronize, load, operate 60 minutes	31 Days	Yes
4.8.1.1.2.b	3.8.1.5	AC Sources – Operating/Remove any accumulated water from day tank	31 Days	Yes
4.8.1.1.2.c	3.8.3.5	Diesel Fuel Oil, Lube Oil, and Starting Air/Remove accumulated water from fuel oil storge tank	31 Days	Yes
4.8.1.1.2.e	N/A	VCSNS TS specific periodic SR for fuel oil sampling.	31 Days	N/A
4.8.1.1.2.f.1	3.8.1.7	AC Sources – Operating/DG fast start	184 Days	Yes
4.8.1.1.2.f.2	N/A	VCSNS TS specific periodic SR for manually synchronizing, loading, and operating the DGs.	184 Days	N/A
4.8.1.1.2.g.2	3.8.1.9	AC Sources – Operating/DG rejects largest load	18 Months	Yes
4.8.1.1.2.g.3	3.8.1.10	AC Sources – Operating/DG does not trip on load rejection	18 Months	Yes
4.8.1.1.2.g.4	3.8.1.11	AC Sources – Operating/DG response to loss of offsite power signal	18 Months	Yes
4.8.1.1.2.g.5	3.8.1.12	AC Sources – Operating/DG response to Engineered Safety Feature signal	18 Months	Yes

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.8.1.1.2.g.6.a and 4.8.1.1.2.g.6.b	3.8.1.19	AC Sources – Operating/DG response to concurrent ESF and loss-of-offsite power signal	18 Months	Yes
4.8.1.1.2.g.6.c	3.8.1.13	AC Sources – Operating/DG bypass of non-critical trips	18 Months	Yes
4.8.1.1.2.g.7	3.8.1.14	AC Sources – Operating/DG operates for 24 hours loaded	18 Months	Yes
4.8.1.1.2.g.8	N/A	VCSNS TS specific periodic SR for verification of auto-connected loads within 2000 hour rating of the DGs.	18 Months	N/A
4.8.1.1.2.g.9	3.8.1.16	AC Sources – Operating/DG synchronize with offsite source and unloads	18 Months	Yes
4.8.1.1.2.g.10	3.8.1.17	AC Sources – Operating/DG responds to ESF actuation while running	18 Months	Yes
4.8.1.1.2.g.11	N/A	VCSNS TS specific periodic SR for verification of fuel transfer via cross-connection lines.	18 Months	N/A
4.8.1.1.2.g.12	3.8.1.18	AC Sources – Operating/Verify load block intervals	18 Months	Yes
4.8.1.1.2.g.13	N/A	VCSNS TS specific periodic SR for verification of DG lockout features.	18 Months	N/A
4.8.1.1.2.g.14	3.8.1.15	AC Sources – Operating/DG starts within 5 minutes after operation	18 Months	Yes
4.8.1.1.2.h	3.8.1.20	AC Sources – Operating/Simultaneous DG start	10 Years	Yes
4.8.1.1.2.i	N/A	See Section 2.2.1, item 2 discussion on 4.8.1.1.2.j.	10 Years	N/A
4.8.2.1.a.1	3.8.6.2 and 3.8.6.3	DC Sources – Operating/verify battery pilot cell voltage (3.8.6.2) and verify battery electrolyte level	7 Days	N/A
4.8.2.1.a.2	3.8.4.1	DC Sources – operating/Verify battery terminal voltage	7 Days	Yes

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.8.2.1.a.2	N/A	VCSNS TS specific periodic SR total battery terminal voltage verification	7 Days	N/A
4.8.2.1.b.1	3.8.6.5	DC Sources – operating/Verify connected cell voltage	92 Days	Yes
4.8.2.1.b.2	N/A	VCSNS TS specific periodic SR (included under 4.8.2.1.b) for check of battery terminal and connecter corrosion or connection resistance.	92 Days	N/A
4.8.2.1.b.3	3.8.6.4	DC Sources – operating/Verify battery pilot cell temperature	92 Days	Yes
4.8.2.1.c.1	N/A	VCSNS TS specific periodic SR (part of 4.8.2.1.c) for check of battery for physical damage or abnormal deterioration.	18 Months	N/A
4.8.2.1.c.2	N/A	VCSNS TS specific periodic SR (part of 4.8.2.1.c) for check of battery for clean, tight connections and anti-corrosion coating on terminals.	18 Months	N/A
4.8.2.1.c.3	N/A	VCSNS TS specific periodic SR (part of 4.8.2.1.c) for check of battery resistances for inter-cell, jumper, and terminal plate connections.	18 Months	N/A
4.8.2.1.c.4	3.8.4.2	DC Sources – operating/Verify battery charger capability	18 Months	Yes
4.8.2.1.d	3.8.4.3	DC Sources – operating/Verify battery capacity	18 Months	Yes
4.8.2.1.e	3.8.6.6	DC Sources – operating/Battery performance discharge test	60 Months	Yes
4.8.3.1	3.8.7.1	Inverters – Operating/Verify voltage and alignment	7 Days	Yes
4.8.3.1	3.8.9.1	Distribution Systems – Operating/Verify voltage and alignment	7 Days	Yes
4.8.3.2	3.8.8.1	Inverters – Shutdown/Verify voltage and alignment	7 Days	Yes

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.8.3.2	3.8.10.1	Distribution Systems – Shutdown/Verify voltage and alignment	7 Days	Yes
4.8.4.1.a	N/A	VCSNS TS specific periodic SR for Containment Penetration Conductor Overcurrent Protective Devices (testing of medium and low voltage breakers).	18 Months	N/A
4.8.4.1.b	N/A	VCSNS TS specific periodic SR for Containment Penetration Conductor Overcurrent Protective Devices (inspection and preventive maintenance of breakers).	60 Months	N/A
4.8.4.3.a	N/A	VCSNS TS specific periodic SR for Circuit Protection Devices (testing of medium and low voltage breakers).	18 Months	N/A
4.8.4.3.b	N/A	VCSNS TS specific periodic SR for Circuit Protection Devices (inspection and preventive maintenance of breakers).	60 Months	N/A
4.9.1.2	3.9.1.1	Boron Concentration/Verify boron concentration	72 Hours	Yes
4.9.1.3	3.9.2.1	Unborated Water Source Isolation Valves/Verify valves secured closed	72 Hours	Yes
4.9.2.a	3.9.3.1	Nuclear Instrumentation/CHANNEL CHECK	12 Hours	Yes
4.9.2.c	N/A	VCSNS TS specific periodic SR for channel operational test of source range neutron flux monitors.	7 Days	N/A
4.9.3.2	N/A	VCSNS TS specific periodic SR for verification of CCW temperature to Spent Fuel Pool Cooling System heat exchanger.	12 Hours	N/A

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.9.5	N/A	VCSNS TS specific periodic SR for verification of direct communications between control room and refueling station during core alterations.	12 Hours	N/A
4.9.7.1.1	3.9.5.1	RHR and Coolant Circulation – High Water Level/Verify RHR loop in operation with required flowrate	12 Hours	Yes
4.9.7.1.2	N/A	VCSNS TS specific SR to verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	31 Days	N/A
4.9.7.2.1	3.9.6.1	RHR and Coolant Circulation – Low Water Level/Verify RHR loop in operation with required flowrate	12 Hours	Yes
4.9.7.2.2	N/A	VCSNS TS specific SR to verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	31 Days	N/A
4.9.9	3.9.7.1	Refueling Cavity Water Level/Verify cavity water level	24 Hours	Yes
4.10.1.1	N/A	VCSNS TS specific periodic SR for determining position of each full-length rod not full inserted during PHYSICS TESTS.	2 Hours	N/A
4.10.2.1	N/A	VCSNS TS specific periodic SR for determining thermal power within limits during PHYSICS TESTS.	1 Hour	N/A
4.10.2.2	N/A	VCSNS TS specific periodic SR for performance of various SRs PHYSICS TESTS.	12 Hours	N/A
4.10.3.1	3.1.8.3	PHYSICS TESTS Exceptions/Verify THERMAL POWER	1 Hour	Yes
4.10.3.3	3.1.8.2	PHYSICS TESTS Exceptions/Verify RCS lowest loop temperature	30 Minutes	Yes

Part 1 - Proposed VCSNS TS Changes vs TSTF-425 Cross-Reference				
VCSNS TS [8.1] SR	**Corresponding SR in NUREG-1431 [8.30]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
4.10.4.1	N/A	VCSNS TS specific periodic SR for determining thermal power within limits during start up and PHYSICS TESTS.	1 Hour	N/A
4.10.5	N/A	VCSNS TS specific periodic SR for rod position indication systems during rod drop time measurements.	24 Hours	N/A
4.11.1.4	N/A	VCSNS TS specific SR to verify the quantity of radioactive material in holdup tanks.	7 Days	N/A
4.11.2.6	N/A	VCSNS TS specific SR to verify the quantity of radioactive material in gas storage tanks.	24 Hours	N/A

Part 2 – List of SRs NUREG-1431 Has and VCSNS TS Does Not Have				
SR in NUREG-1431 [8.30]	**SR in VCSNS TS [8.1]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
3.1.6.3	VCSNS TS does not have this SR	Control Bank Insertion Limits/Verify sequence and overlap limits for control banks.	N/A	Yes
3.1.8.4	VCSNS TS does not have this SR	Control Bank Insertion Limits/Verify SDM.	N/A	Yes
3.2.1.2	VCSNS TS does not have this SR	$F_Q(Z)$/Verify $F_Q^W(Z)$ in limit	N/A	Yes
3.3.4.2	VCSNS TS does not have this SR	Remote Shutdown System/Control circuit and transfer switch verification	N/A	Yes
3.3.4.4	VCSNS TS does not have this SR	Remote Shutdown System/Trip Actuating Device Operational Test (TADOT) of reactor trip breaker indication	N/A	Yes
3.3.5	VCSNS TS does not have separate section for this function.	Loss of Power (LOP) DG Start Instrumentation	N/A	Yes
3.3.5.1	VCSNS TS does not have this SR	Loss of Power (LOP) DG Start Instrumentation/CHANNEL CHECK	N/A	Yes

Part 2 – List of SRs NUREG-1431 Has and VCSNS TS Does Not Have				
SR in NUREG-1431 [8.30]	**SR in VCSNS TS [8.1]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
3.3.6	VCSNS TS does not have separate section for this function	Containment Purge and Exhaust Isolation Instrumentation	N/A	Yes
3.3.7	VCSNS TS does not have separate section for this function	Control Room Emergency Filtration System (CREFS) Actuation Instrumentation	N/A	Yes
3.3.8	VCSNS TS does not have separate section for this function	FBACS Actuation Instrumentation	N/A	Yes
3.3.9	VCSNS TS does not have this TS	BDPS	N/A	Yes
3.4.7.3	VCSNS TS does not have this SR	RCS Loops – MODE 5, Loops Filled/Verify breaker alignment and indicated power	N/A	Yes
3.4.8.2	VCSNS TS does not have this SR	RCS Loops – MODE 5, Loops Not Filled/Verify breaker alignment and indicated power	N/A	Yes
3.4.9.3	VCSNS TS does not have this SR	Pressurizer/Verify pressurizer heater emergency power	N/A	Yes
3.4.11.3	VCSNS TS does not have this SR	Pressurizer PORVs/Cycle each solenoid and accumulator check valve	N/A	Yes
3.4.11.4	VCSNS TS does not have this SR	Pressurizer PORVs/Cycle PORVs and block valve emergency power	N/A	Yes
3.4.12.3	VCSNS TS does not have this SR	LTOP System/Verify each accumulator isolated	N/A	Yes
3.4.12.6	VCSNS TS does not have this SR	VCSNS does not have PORVs for LTOP	N/A	Yes
3.4.12.7	VCSNS TS does not have this SR	LTOP System/Verify RHR isolation valve locked open	N/A	Yes
3.4.12.8	VCSNS TS does not have this SR	VCSNS does not use PORVs for LTOP	N/A	Yes
3.4.12.9	VCSNS TS does not have this SR	VCSNS does not use PORVs for LTOP	N/A	Yes
3.4.14.3	VCSNS TS does not have this SR	RCS PIV Leakage/Verify RHR System autoclosure interlock closes valves	N/A	Yes

Part 2 – List of SRs NUREG-1431 Has and VCSNS TS Does Not Have				
SR in NUREG-1431 [8.30]	**SR in VCSNS TS [8.1]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
3.4.17.1	VCSNS TS does not have this SR	RCS Loop Isolation Valves/Verify each valve open with power removed	N/A	Yes
3.4.19.1	VCSNS TS does not have this SR	RCS Loops – Test Exceptions/Verify THERMAL POWER is < P-7	N/A	Yes
3.5.5	VCSNS TS does not have this section for this function	Seal Injection Flow	N/A	Yes
3.5.6	VCSNS TS does not have BIT	BIT	N/A	Yes
3.6.3.6	VCSNS TS does not have this SR	Containment isolation Valves/Cycle check valves (testable during operation)	N/A	Yes
3.6.3.9	VCSNS TS Does not have this SR	Containment Isolation Valves/Cycle check valves (not testable during operation)	N/A	Yes
3.6.3.10	VCSNS TS does not have this SR	Containment Isolation Valves/Verify purge valve opening limit	N/A	Yes
3.6.8	VCSNS TS Does not have a Shield Building	Shield Building	N/A	Yes
3.6.9	VCSNS TS does not have SR related to this function	Hydrogen Mixing	N/A	Yes
3.6.10	VCSNS TS does not have this system	Hydrogen Igniter System	N/A	Yes
3.6.13	VCSNS TS does not have this system	Shield Building Air Cleanup System	N/A	Yes
3.6.14 – 3.6.18	VCSNS TS does not have ice condenser containment	Ice Condenser Sub-systems	N/A	Yes
3.7.2.2	VCSNS TS does not have this SR	MSIVs/Verify MSIV actuation	N/A	Yes

Part 2 – List of SRs NUREG-1431 Has and VCSNS TS Does Not Have				
SR in NUREG-1431 [8.30]	**SR in VCSNS TS [8.1]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
3.7.3.2	VCSNS TS does not have this SR	MFIVs and MFRVs/Verify valve actuation	N/A	Yes
3.7.4.1	VCSNS TS does not have this SR	ADVs/Cycle each ADV	N/A	Yes
3.7.4.2	VCSNS TS does not have this SR	ADVs/Cycle each ADV block valve	N/A	Yes
3.7.7.2	VCSNS TS does not have this SR	Component Cooling Water (CCW) System/Verify automatic valve actuation	N/A	Yes
3.7.7.3	VCSNS TS does not have this SR	Component Cooling Water (CCW) System/Verify automatic pump start	N/A	Yes
3.7.8.2	VCSNS TS does not have this SR	Service Water System (SWS)/Verify automatic valve actuation	N/A	Yes
3.7.8.3	VCSNS TS does not have this SR	Service Water System (SWS)/Verify automatic pump start	N/A	Yes
3.7.9.3	VCSNS TS does not have this SR	Ultimate Heat Sink (UHS)/Operate each cooling tower fan 15 minutes	N/A	Yes
3.7.9.4	VCSNS TS does not have this SR	Ultimate Heat Sink (UHS)/Verify automatic start of cooling tower fans	N/A	Yes
3.7.10.4	VCSNS TS does not have this SR	Control Room Emergency Filtration System (CREFS)/Verify one CREFS train maintains positive pressure	N/A	Yes
3.7.11	VCSNS TS does not have this section	Control Room Emergency Air Temperature Control System (CREATCS)	N/A	Yes
3.7.12	VCSNS TS does not have this section	ECCS PREACS	N/A	Yes
3.7.13	VCSNS TS does not have this section	FBACS	N/A	Yes
3.7.14	VCSNS TS does not have this section	PREACS	N/A	Yes

Part 2 – List of SRs NUREG-1431 Has and VCSNS TS Does Not Have				
SR in NUREG-1431 [8.30]	**SR in VCSNS TS [8.1]**	**TSTF-425 Section Title/SR Description**	**VCSNS TS [8.1] Surveillance Frequency**	**SR Frequency modified by TSTF-425 [8.3]**
3.8.1.8	VCSNS TS does not have this SR	AC Sources – Operating/Verify transfer from normal to alternate offsite source	N/A	Yes
3.8.3	VCSNS TS does not have this section	Diesel Fuel Oil, Lube Oil, and Starting Air	N/A	Yes
3.8.3.2	VCSNS TS does not have this SR	Diesel Fuel Oil, Lube Oil, and Starting Air/Verify lube oil inventory	N/A	Yes
3.8.3.4	VCSNS TS does not have this SR	Diesel Fuel Oil, Lube Oil, and Starting Air/Verify DG air start receiver pressure	N/A	Yes
3.8.6	VCSNS TS does not have this section	Battery Parameters	N/A	Yes
3.8.6.1	VCSNS TS does not have this SR	Battery Parameters/Verify battery float current	N/A	Yes
3.8.9	VCSNS TS does not have this SR	Distribution Systems - Operating	N/A	Yes
3.8.9	VCSNS TS does not have this section	N/A ~ Electrical Power Systems/Distribution Systems	N/A	Yes
3.9.2	VCSNS TS does not have this section	Unborated Water Source Isolation Valves	N/A	Yes
3.9.3.2	VCSNS TS does not have this SR	Nuclear Instrumentation/CHANNEL CALIBRATION	N/A	Yes
3.9.4	VCSNS TS does not have this section	Containment Penetrations	N/A	Yes
3.9.6.2	VCSNS TS does not have this SR	RHR and Coolant Circulation – Low Water Level/verify breaker alignment and power available	N/A	Yes

ATTACHMENT 8

REFERENCES

8.1 VCSNS Technical Specifications

8.2 CM-AA-STI-101, "Risk Informed Technical Specifications Surveillance Frequency Control Program"

8.3 Technical Specification Task Force Improved Standard Technical Specifications Change Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b," dated March 18, 2009.

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

8.5 Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies" (ML071360456), Revision 1, April 20007

8.6 Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Revision 2, March 2009

8.7 USNRC, Notification of Issue with NRC-Approved Technical Specifications Task Force (TSTF) Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b," April 14, 2010 (ADAMS Accession No. ML 100990099.)

8.8 10 CFR 50.91(a), Notice for Public Comment

8.9 ML 14023A748, Millstone Power Station, Unit No.3 - Issuance of Amendment Re: Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

8.10 NOTEBK-PRA-VCS-QU.2, Revision 0, "Quantification Results," (Dominion Energy Services)

8.11 ML19199A696, Letter from Mary Jane Ross-Lee to Daniel G. Stoddard, "Virgil C. Summer Nuclear Station, Unit 1 -Staff Review of Seismic Probabilistic Risk Assessment Associated with Reevaluated Seismic Hazard Implementation of The Near-Term Task Force Recommendation 2.1: Seismic (EPID NO. L-2018-JLD-0013)"

8.12 ML19305A005, Letter from Michael T. Markley to Daniel G. Stoddard, "Virgil C. Summer Nuclear Station, Unit No. 1, Issuance of Amendment No. 217, Re: National Fire Protection Association Standard (NFPA) 805 Program Revisions (EPID L-2018-LLA-0233)"

8.13 NOTEBK-PRA-VCS-RA.000, Revision 0, "V. C. Summer PRA Day 0 Quantification," (Dominion Energy Services)

8.14 Serial No. 19-339B, South Carolina Electric & Gas Company, Virgil C. Summer Nuclear Station (VCSNS) Unit 1, License Amendment Request- LAR-16-01490, NFPA 805 Program Revisions Response to Follow-Up Question to PRA RAI 03

8.15 Letter from Karen Cotton, U.S. NRC to Stephen Byrne, SCE&G. "Subject: Review of Virgil C. Summer Individual Plant Examination of External Events (IPEEE) Submittal (TAC NO. M83680)," dated June 14, 2000

8.16 ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications

8.17 PWROG-16051-P, "Peer Review of the V. C. Summer Nuclear Station Internal Events and Internal Flood Probabilistic Risk Assessment," dated February 2017

8.18 ASME/ANS RA-Sb-2013, Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications

8.19 NEI 05-04, Rev. 2, Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard

8.20 NEI 07-12, Rev. 1, Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines

8.21 NEI 12-13, External Hazards PRA Peer Review Process Guidelines

8.22 LTR-RAM-II-10-068, "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 VC Summer Fire Probabilistic Risk Assessment," December 2010

8.23 LTR-RAM-II-11-025, "Fire PRA Follow-on 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 VC Summer Nuclear Station Fire Probabilistic Risk Assessment," dated July 2011

8.24 RG 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis

8.25 ML14287A289, Letter, Shawn Williams, NRC, to Thomas Gatlin, "Virgil C. Summer Nuclear Station, Unit 1 – Issuance of Amendment Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c) (TAC No. ME7586)", dated February 11, 2015.

8.26 ML14311A128, "Final VC Summer FPRA ROR," Revision 1, dated October10, 2014.

8.27 05093.001-RPT-13383, "AP1000 V.C. Summer 2 & 3 At-Power Seismic PRA 2017 Peer Review Using ASME/ANS PRA Standard Requirements," dated November 6, 2017

8.28 PWROG-18037-P, "Peer Review of the V.C. Summer Unit 1 Seismic Probabilistic Risk Assessment," dated July 2018

8.29 PWROG-18050-P, "Independent Assessment of Facts & Observations Closure of the V.C. Summer Unit 1 Seismic Probabilistic Risk Assessment," dated August 2018

8.30 NUREG-1431, Standard Technical Specifications, Westinghouse Plants

8.31 10 CFR 50.92(c), Issuance of Amendment

8.32 Regulatory Guide 1.177, An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications

8.33 Regulatory Guide 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3, dated December 2020

8.34 NUREG-0452, Revision 2, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors"

8.35 P3586-01-03, "Focused Scope Peer Review of the V.C. Summer (VCS) Internal Events PRA," dated December 2020