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January 25, 2012

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No.	11-687
NSSL/MLC	R0
Docket No.	50-336
License No.	DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
LICENSE AMENDMENT REQUEST TO RELOCATE TS SURVEILLANCE FREQUENCIES TO LICENSEE CONTROLLED PROGRAM IN ACCORDANCE WITH TSTF-425, REVISION 3

In accordance with the provisions of 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) is submitting a request for an amendment to the technical specifications (TS) for Millstone Power Station Unit 2 (MPS2). The proposed amendment would modify TSs by relocating specific surveillance frequencies to a licensee-controlled program with the adoption of Technical Specification Task Force (TSTF)–425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – Risk-Informed Technical Specification Task Force (RITSTF) Initiative 5b." Additionally, the change would add a new program, the Surveillance Frequency Control Program (SFCP), to TS Section 6, Administrative Controls. The changes are consistent with NRC-approved Industry/TSTF Standard Technical Specifications (STS) change TSTF-425, Revision 3, (ADAMS Accession No. ML090850642). The Federal Register notice published on July 6, 2009 (74 FR 31996), announced the availability of this TS improvement.

Attachment 1 provides a description and assessment of the proposed change. Attachment 2 includes DNC documentation with regard to Probabilistic Risk Assessment technical adequacy. Attachment 4 provides a cross-reference between the NUREG-1432 surveillances included in TSTF-425 versus the MPS2 surveillances included in this amendment request. Attachments 3 and 6 provide the MPS2 marked-up TS pages and TS Bases pages, respectively. The marked-up TS Bases pages are provided for information only. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program upon approval of this amendment request.

As detailed in Attachment 5, the proposed amendment does not involve a Significant Hazards Consideration pursuant to the provisions of 10 CFR 50.92. The Facility Safety Review Committee has reviewed and concurred with the determinations herein.

Issuance of this amendment is requested no later than January 28, 2013 with the amendment to be implemented within 60 days.

A001
NRR

In accordance with 10 CFR 50.91(b), a copy of this license amendment request is being provided to the State of Connecticut.

Should you have any questions in regard to this submittal, please contact Wanda Craft at (804) 273-4687.

Sincerely,

J. Alan Price
Vice President – Nuclear Engineering

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2014

COMMONWEALTH OF VIRGINIA
COUNTY OF HENRICO

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price, who is Vice President – Nuclear Engineering of Dominion Nuclear Connecticut, 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 25TH day of Jan, 2012.

My Commission Expires: May 31, 2014

Vicki L. Hull
Notary Public

Attachments:

1. Description and Assessment of Proposed Changes
2. Documentation of PRA Technical Adequacy
3. Marked-up Technical Specifications Changes
4. Cross-References - NUREG-1432 to MPS2 TS Surveillance Frequencies Removed
5. Significant Hazards Consideration Determination
6. Marked-Up Technical Specifications Bases Changes (For Information Only)

Commitments made in this letter: None

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ATTACHMENT 1

Description and Assessment of Proposed Changes

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

DESCRIPTION AND ASSESSMENT OF PROPOSED CHANGES

1.0 DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) is submitting a request for an amendment to the technical specifications (TSs) for Millstone Power Station Unit 2 (MPS2). The proposed amendment would modify TSs by relocating specific surveillance frequencies to a licensee-controlled program with the adoption of Technical Specification Task Force (TSTF)–425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b." Additionally, the change would add a new program, the Surveillance Frequency Control Program (SFCP), to TS Section 6, Administrative Controls. The changes are consistent with NRC-approved Industry/TSTF Standard Technical Specifications (STS) change TSTF-425, Revision 3, (ADAMS Accession No. ML090850642). The Federal Register notice published on July 6, 2009 (74 FR 31996), announced the availability of this TS improvement.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

DNC has reviewed the safety evaluation provided in Federal Register Notice 74 FR 31996, dated July 6, 2009. This review included a review of the NRC staff's evaluation, TSTF-425, Revision 3, and the requirements specified in NEI 04-10, Rev. 1 (ADAMS Accession No. ML071360456).

Attachment 2 includes DNC documentation with regard to the technical adequacy of the probabilistic risk assessment (PRA) consistent with the requirements of Regulatory Guide (RG) 1.200, Revision 1 (ADAMS Accession No. ML070240001), Section 4.2. Attachment 2 also describes any PRA models without NRC-endorsed standards, including documentation of the quality characteristics of those models in accordance with RG 1. 200.

DNC has concluded that the justifications presented in the TSTF proposal and the safety evaluation prepared by the NRC staff are applicable to MPS2 and justify this amendment to incorporate the changes to the MPS2 TSs.

2.2 Optional Changes and Variations

The proposed amendment is consistent with the STS changes described in TSTF-425, Revision 3. However, DNC proposes variations or deviations from TSTF-425, as identified below.

1. Revised (typed) TS pages are not included in this amendment request given the number of TS pages affected, the straightforward nature of the proposed changes, and outstanding MPS2 amendment requests that may impact some of the same TS pages. Providing only mark-ups of the proposed TS changes satisfies the requirements of 10 CFR 50.90 in that the mark-ups fully describe the changes desired. This represents an administrative deviation from the NRC staff's model application dated July 6, 2009 (74 FR 31996) with no impact on the NRC staff's model safety evaluation published in the same Federal Register Notice. As a result of this deviation, the contents and numbering of the attachments for this amendment request differ from the attachments specified in the NRC staff's model application. The proposed TS Bases changes are provided to the NRC for information.

2. The inserts provided in TSTF-425 are revised to fit the MPS2 TS format.

   The TSTF-425 insert for each relocated surveillance frequency is changed from "in accordance with the Surveillance Frequency Control Program to "at the frequency specified in the Surveillance Frequency Control Program."

   The insert provided in TSTF-425 to replace text describing the basis for each frequency relocated to the SFCP has been revised from "The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program" to read "The(se) Surveillance Frequency(ies) is/are controlled under the Surveillance Frequency Control Program." This deviation is consistent with recent NRC guidance. After NRC approval of the license amendment request (LAR) and as part of the LAR implementation, the existing MPS2 Bases information describing the basis for the relocated surveillance frequencies will also be relocated to a licensee-controlled program with the relocated surveillance frequencies.

   In addition, other editorial changes to the existing TS wording and/or text inserts are being made. These administrative/editorial deviations of the TSTF-425 inserts and the existing TS wording are necessary to fit the MPS2 TS format.

3. Attachment 4 provides a cross-reference between the NUREG-1432 surveillances included in TSTF-425 versus the MPS2 surveillances included in this amendment request. Attachment 4 includes a summary description of the referenced TSTF-425 (NUREG-1432)/MPS2 TS surveillances which is provided for information purposes only and is not intended to be a verbatim description of the TS surveillances. This cross reference highlights the following:

   a. NUREG-1432 surveillances included in TSTF-425 and corresponding MPS2 surveillances with plant-specific surveillance numbers,

b. NUREG-1432 surveillances included in TSTF-425 that are not contained in the MPS2 TS, and

c. MPS2 plant-specific surveillances that are not contained in NUREG-1432 and, therefore, are not included in the TSTF-425 mark-ups.

Since the MPS2 TSs are custom TSs, the applicable surveillance requirements and associated Bases numbers differ from the STSs presented in NUREG-1432 and TSTF-425, but with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

For NUREG-1432 surveillances not contained in MPS2 TSs, the corresponding mark-ups included in TSTF-425 for these surveillances are not applicable to MPS2. This is an administrative deviation from TSTF-425 with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

For MPS2 plant-specific surveillances not included in the NUREG-1432 markups provided in TSTF-425, DNC has determined that since these surveillances involve fixed periodic frequencies, relocation of these frequencies is consistent with TSTF-425, Revision 3, and with the NRC's model safety evaluation dated July 6, 2009 (74 FR 31996), including the scope exclusions identified in Section 1.0, "Introduction," of the model safety evaluation. In accordance with TSTF-425, changes to the frequencies for these surveillances would be controlled under the SFCP.

There are several instances in the MPS2 TSs where the words 'and' and 'or' appear at the end of a surveillance requirement. In most cases, these words are not intended to be logical connectors which place the constraints of the preceding surveillance requirement (often times event-driven) on the remaining portion of the surveillance but rather are used for purposes of readability and flow. This situation applies to the following SRs: 4.1.1.2, 4.1.1.5b, 4.1.3.1.1, 4.1.3.1.4b, 4.2.3.2b, 4.5.1d, 4.9.16.1 and 4.9.17.

As currently written, SR 4.2.1.3b does not specify a surveillance frequency, however; it is performed at least once per 31 days, as required by its applicable station surveillance procedure. As a result, the markup for this SR references the SFCP in accordance with TSTF-425.

The SFCP provides the necessary administrative controls to require that surveillances related to testing, calibration, and inspection are conducted at a frequency to assure the necessary quality of systems and components is maintained, facility operation will be within safety limits, and the limiting conditions for operation will be met. Changes to frequencies in the SFCP would be evaluated using the methodology and PRA guidelines contained in NEI 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," as approved by NRC letter dated September 19, 2007 (ADAMS Accession No. ML072570267). The NEI 04-10, Revision 1 methodology includes qualitative considerations, risk analyses, sensitivity studies and bounding analyses, as necessary, and recommended monitoring of the performance of structures, systems, and components (SSCs) for which frequencies are changed to assure that reduced testing does not adversely impact the SSCs. In addition, the NEI 04-10, Revision 1 methodology satisfies the five key safety principles specified in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998, relative to changes in surveillance frequencies.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration

DNC has reviewed the proposed no significant hazards consideration (NSHC) determination published in the Federal Register dated July 6, 2009 (74 FR 31996). DNC has concluded that the proposed NSHC presented in the Federal Register notice is applicable to MPS2, and is provided as Attachment 5 to this amendment request, which satisfies the requirements of 10 CFR 50.91 (a).

3.2 Applicable Regulatory Requirements

A description of the proposed changes and their relationship to applicable regulatory requirements is provided in TSTF-425, Revision 3 and the NRC's model safety evaluation published in the Notice of Availability dated July 6, 2009 (74 FR 31996). DNC has concluded that the relationship of the proposed changes to the applicable regulatory requirements presented in the Federal Register notice is applicable to MPS2.

3.3 Conclusions

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

4.0 ENVIRONMENTAL CONSIDERATION

DNC has reviewed the environmental consideration included in the NRC staff's model safety evaluation published in the Federal Register on July 6, 2009 (74 FR 31996). DNC has concluded that the staff's findings presented therein are applicable to MPS2, and the determination is hereby incorporated by reference for this application.

5.0 REFERENCES

1. TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b," March 18, 2009 (ADAMS Accession Number: ML090850642).

2. 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).

3. NEI 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," April 2007 (ADAMS Accession Number: ML071360456).

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

5. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998 (ADAMS Accession No. ML003740176).

ATTACHMENT 2

Documentation of Probabilistic Risk Assessment (PRA) Technical Adequacy

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

Documentation of Probabilistic Risk Assessment (PRA) Technical Adequacy

1.0 PURPOSE

The purpose of this risk assessment is to provide the Probabilistic Risk Assessment (PRA) technical adequacy of the Millstone Power Station Unit 2 (MPS2) model, M209Aa, to support the Risk-Informed Technical Specification Initiative (RITS) 5b. This includes status of critical PRA model reviews during the PRA Peer Review and a gap assessment with respect to American Society of Mechanical Engineers (ASME) PRA Standard RA-Sb-2005 and its endorsing Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.200, Rev. 1.

2.0 INTRODUCTION

The MPS2 PRA model has benefited from the comprehensive technical PRA peer review and self-assessment. These include the MPS2 internal events PRA receiving a formal industry PRA Peer Review in 1999 (Ref. 6.1) and a self-assessment/independent review of the MPS2 PRA against Addendum B of the ASME/ANS PRA Standard and RG 1.200, Revision 1 (Ref. 6.3).

3.0 ANALYSIS

Documentation of the PRA technical adequacy includes the following information:

1. Proposed Risk-Informed Application
    - Description of RITS 5b process
2. PRA Quality Overview
3. Technical Adequacy of the PRA Model
    - PRA Maintenance and Update
    - PRA Model timeline of improvements
4. Comprehensive Critical Reviews
    - CEOG PRA Peer Review
    - MPS2 PRA Self-Assessment
5. Status of Identified Gaps to NEI 00-02 and Capability Category II of the ASME PRA Standard
6. External Events Considerations
    - Fire Risk
    - Seismic Risk
    - High Winds, Floods and Other External Events
7. Summary

3.1 Proposed Risk-Informed Application

The implementation of the Surveillance Frequency Control Program (SFCP, also referred to as RITS 5b) at MPS2 will follow the guidance provided in NEI 04-10, Revision 1 (Ref. 6.5) 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 STI revision is reviewed to determine whether there are any commitments made to the NRC that may prohibit changing the interval. If there are no related commitments, or the commitments may be changed using a commitment change process based on NRC endorsed guidance, then evaluation of the STI revision would proceed. If a commitment exists and the commitment change process does not permit the change, then the STI revision would not be implemented. Only after receiving formal NRC approval to change the commitment would a STI revision proceed.

- A qualitative analysis is performed for each STI revision that involves several considerations as explained in NEI 04-10, Revision 1.

- Each STI revision is reviewed by an expert panel, referred to as the Integrated Decision-making Panel (IDP), which is normally the same panel used for Maintenance Rule implementation, but with the addition of specialists with experience in surveillance tests and system or component reliability. If the IDP approves the STI revision, the change is documented 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. The performance monitoring helps to confirm that no failure mechanisms related to the revised test interval become important enough to alter the information provided for the justification of the interval changes.

- The IDP is responsible for periodic review of performance monitoring results. If it is determined that the time interval between successive performances of a surveillance test is a factor in the unsatisfactory performances of the surveillance, the IDP returns the STI to the previously acceptable STI.

- In addition to the above steps, the PRA is used, when possible, to quantify the effect of a proposed individual STI revision compared to acceptance criteria in NEI 04-10, Revision 1. Also, the cumulative impact of all risk-informed STI revisions on all PRA evaluations (i.e., internal events, external events and shutdown) is also compared to the risk acceptance criteria as delineated in NEI 04-10, Revision 1. For those cases where the STI cannot be modeled in the plant PRA, or where a particular PRA model does not exist for a given hazard group, a qualitative or bounding analysis is performed to provide justification for the acceptability of the proposed test interval change.

3.2 PRA Quality Overview

The NEI 04-10, Revision 1 methodology endorses the guidance provided in RG 1.200, Revision 1 (Ref. 6.7), "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The guidance in RG 1.200 indicates that the following steps should be followed when performing PRA assessments:

1. Identify the parts of the PRA used to support the application.
   - Structures, systems, and components (SSCs), operational characteristics affected by the application and how these are implemented in the PRA model.
   - A definition of the acceptance criteria used for the application.

2. Identify the scope of risk contributors addressed by the PRA model.
   - If not full scope (i.e., internal events, external events, shutdown), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the PRA model.

3. Summarize the risk assessment methodology used to assess the risk of the application.
   - Include how the PRA model was modified to appropriately model the risk impact of the change request.

4. Demonstrate the technical adequacy of the PRA.
   - Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.
   - Document peer review findings and observations that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed, justify why the significant contributors would not be impacted.

- Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the RG (specifically RG 1.200, Revision 1, which includes only internal events). Provide justification to show that where specific requirements in the standard are not adequately met, it will not unduly impact the results.
- Identify key assumptions and approximations relevant to the results used in the decision-making process.

(NOTE: Because of the broad scope of potential Initiative 5b applications, and the fact that the risk assessment details will differ from application to application, each of the issues encompassed in Items 1 through 3 above will be covered with the preparation of each individual PRA assessment made in support of the individual STI interval requests. Item 3 satisfies one of the requirements of Section 4.2 of RG 1.200. The remaining requirements of Section 4.2 are addressed by Item 4, which is described in the next section.)

3.3 Technical Adequacy of the PRA Model

Dominion employs a structured approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Dominion nuclear generating sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the MPS2 PRA.

<u>PRA Maintenance and Update</u>

The MPS2 PRA model of record, M209Aa, and associated documentation, has been maintained as a living program and the PRA is updated approximately every 3 to 5 years to reflect the as-built, as-operated plant. The M209Aa PRA model is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the MPS2 PRA is based on the event tree/fault tree methodology, which is a well-known methodology in the industry.

There are several procedures and GARDs (Guidance and Reference Documentation) that govern Dominion'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 process (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 the Dominion Nuclear Fleet, including Kewaunee, Millstone, North Anna and Surry Power Stations
- 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

The purpose of the NF-AA-PRA GARDs and Procedures is to provide information and guidelines for performing PRA. Nevertheless, non-routine risk assessments are often unique, requiring departure from these guidelines and information in order to correctly perform and meet the risk assessment objectives. Such departure must be evaluated and documented in accordance with applicable regulations and Dominion policies.

An administratively controlled process is used to maintain configuration control of the MPS2 PRA models, data, and software. In addition to model control, administrative mechanisms are in place to assure that plant modifications, procedure changes, system operation changes and industry operating experience (OE) are appropriately screened, dispositioned and scheduled for incorporation into the model. These processes help assure that the MPS2 PRA reflects the as-built, as-operated plant within the limitations of the PRA methodology.

The process for performing PRA involves a periodic review and update cycle to model any changes in the plant design or operation. Plant hardware and procedure changes are reviewed on an approximate quarterly or more frequent basis to determine if they impact the PRA and if a PRA model 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 expert panel. If a non-trivial impact is expected, then this may include the performance of additional sensitivity studies or PRA model changes to confirm the impact on the risk analysis.

The Level 1 and Level 2 MPS2 PRA analyses were originally developed and submitted to the NRC in 1993 as the Individual Plant Examination (IPE) Summary Report (Ref. 6.10). In response to Supplement 4 of Generic Letter 88-20, the IPE External Events (IPEEE) Summary Report was submitted to the NRC in 1995 (Ref. 6.11). The MPS2 PRA has been updated many times since the original IPE.

Since 1995, updates have been made to incorporate plant and procedure changes, update plant-specific reliability and unavailability data, improve the fidelity of the model, incorporate Combustion Engineering Owners Group (CEOG) Peer Review comments and support other applications, such as On-line Maintenance, Risk-Informed In-Service Inspection (RI-ISI), Maintenance Rule Risk Significance, and Mitigating System Performance Index (MSPI).

The enhancements to the MPS2 PRA model include a major internal flooding update and number of updates to the Level 2 PRA model to allow a more realistic assessment of the Large Early Release Frequency (LERF). A summary of the MPS2 PRA history is listed below.

Date	Model Change
12/93	IPE submitted
05/94	Supplement regarding a potential vulnerability identified in IPE submittal
09/95	Responses to RAIs on the IPE submittal provided
12/95	IPEEE submitted
05/96	IPE approved by NRC
11/99	CEOG peer review report completed
01/00	PRA model updated - Plant-specific data incorporated
06/00	PRA model updated - Addressed significant peer review comments
01/01	IPEEE approved by NRC
04/01	PRA model updated - Incorporated design change to electrically separate from Unit 1 and connect to Unit 3
12/05	PRA model updated - Plant-specific data incorporated
10/07	Initial PRA self-assessment performed
01/11	PRA model updated - Addressed not met ASME/ANS supporting requirements
02/11	Updated PRA self-assessment based on latest PRA model and regulatory requirements

3.4 Comprehensive Critical Reviews

The MPS2 PRA model has benefited from the comprehensive technical PRA Peer Reviews:

CEOG PRA Peer Review

The MPS2 internal events PRA received a formal industry PRA Peer Review in 1999 (Ref. 6.1). The purpose of the PRA Peer Review process was to provide a method for establishing the technical quality of a PRA for the spectrum of potential risk-informed plant licensing applications for which the PRA may be used. The PRA Peer Review process used a team composed of industry PRA and system analysts, each with significant expertise in both PRA development and PRA applications. This team provided both an objective review of the PRA technical elements and a subjective assessment, based on their PRA experience, regarding the acceptability of the PRA elements. The team used a set of checklists as a framework within which to evaluate the scope, comprehensiveness, completeness, and fidelity of the PRA products available. The MPS2 review team used the NEI-00-02 "PRA Peer Review Process Guidance" as the basis for the review.

The general scope of the implementation of the PRA Peer Review included review of eleven main technical elements, using checklist tables (to cover the elements and sub-elements), for an at-power PRA including internal events, internal flooding, and containment performance (with focus on LERF).

The findings and observations from the PRA Peer Review were prioritized into four categories (A through D) based upon importance to the completeness of the model. With the exception of one Category B comment, all comments in Categories A and B have been addressed. The remaining Category B comment is listed in Section 3.5.

MPS2 PRA Self Assessment

Reference 6.3 documents the results of a self assessment/independent review of the MPS2 PRA model, data, and documentation in accordance with the Capability Category II requirements of the ASME Standard for PRA (Ref. 6.6) and RG 1.200 (Ref. 6.7). The initial review was performed by Dominion in 2007 with support from a contracting company, MARACOR, using a team of experts with experience in performing NEI PRA Certifications and ASME PRA Standard Reviews. The assessment included a review of the Dominion PRA procedures, current documentation notebooks, and other documentation.

The intent of this independent assessment was to provide a basic assessment of the current PRA against the ASME standard and the RG to determine if each of the requirements of Capability Category II had been met and documented. The assessment team reviewed the technical adequacy of compliance with each of the requirements as compared to current PRA practices in the industry. Insights gained from recent industry programs to comply with the ASME standard were also used.

All technical areas, described in Section 4 of the ASME standard and RG 1.200, have been reviewed, with the exception of the PRA Configuration Control Program. During this review, specific "Facts and Observations" (F&Os) were not generated. However, specific recommendations were provided for each supporting requirement, which was assessed as not met by the current PRA model and documentation. These recommendations were entered into the PRACC database and will be used directly to guide future PRA enhancement activities. The PRACC database is being used to track each supporting requirement that was assessed as not met in a corresponding database item. Of the 328 supporting requirements, the MPS2 PRA does not meet 39 Category II supporting requirements. Section 3.5 lists the supporting requirements not met after the M209Aa model update. The self-assessment was performed against the previous ASME standard (Ref. 6.7), but Section 3.5 lists the supporting requirement numbers from the current ASME/ANS standard (Ref. 6.12).

3.5 Identify Gaps between PRA Model and applicable PRA standard

References 6.2 and 6.3 contain the gap analysis between the PRA capability and PRA standards (i.e., CEOG peer review and ASME standards). There are 39 ASME standard supporting requirements not met and one peer review element not met. Of the 40 total elements not met, 14 could impact the RITS 5b application while the remaining 26 pertain only to documentation requirements. Table 1 groups these 14 not met supporting requirements into eight categories and evaluates the impact of the gap on the RITS 5b application. If the gap potentially affects components that could be subject to the RITS 5b application, then a sensitivity study will be performed as part of the surveillance frequency change evaluation. Table 2 lists the gaps and provides an assessment of the potential impact on implementation of the SFCP or RITS 5b.

It is important to note that for each element in the ASME PRA Standard there is a separate high level requirement for documentation. Dominion made the decision in order to meet Category II for a supporting requirement, there had to be documented evidence that the supporting requirement was met. Since each high level requirement of the standard has a separate documentation part, the supporting requirement could have been categorized as met with the documentation part categorized as not met. Dominion's approach was to conservatively categorize the supporting requirement as not met due to documentation issues. Therefore, there are numerous technical supporting requirements that are "not met" for lack of documentation. For example, IE-A6 is not met due to the lack of documented evidence for plant personnel interviews. Dominion agrees that documentation is essential in maintaining PRAs and understanding the results.

Table 1 - Justification for Gap not Impacting RITS 5b Application			
Element Not Met	**Element Description**	**Review Comment**	**Importance to Application**
IE-C1b [IE-C3]	CREDIT recovery actions [those implied in IE-C4(c), and those implied and discussed in IE-C6 through IE-C9] as appropriate. JUSTIFY each such credit (as evidenced such as through procedures or training).	Recovery actions are appropriately credited in the initiating event (IE) analysis and each such credit is justified (all credited actions are proceduralized). However, the station blackout (SBO) initiating event fault tree logic includes the potential to align to MPS3 power transformers or the SBO diesel. Such actions would occur after the SBO initiating event (available response times for the actions are approximately 100 minutes) and would appear to be more appropriately modeled in the post-initiator portions of the SBO logic (e.g., the power recovery function).	As part of the 2009 model update, the IE series notebooks were revised to address the supporting requirements not met in the self assessment. The SBO model changes recommended in the self assessment for this supporting requirement will not be made because the SBO accident sequence development would not change if a separate node was added to the SBO event tree to include starting aligning the SBO diesel or power from the other unit. This gap has no impact on the RITS 5b application.
AS-10	Dependencies among top events are identified and addressed.	Main Feedwater success criteria do not require makeup to the condenser when steam dump valves fail. No documentation of the verification that adequate volume exists in the condenser for successful cooldown. No modeling of makeup to the condenser was identified.	Given that there are four steam dump valves with only one valve required to provide adequate condenser inventory and the main feedwater pumps rely on the same support systems as the steam dump valves (i.e., Instrument Air (IA) and Main Condenser), the impact of adding the steam dump valves as a required support system for Main Feedwater has an insignificant impact on the overall model. The steam dump valves are not required by technical specifications and are therefore, not in-scope to the RITS 5b process. Consequently, this gap has no impact on the RITS 5b application.
AS-A7	DELINEATE the possible accident sequences for each modeled initiating event, unless the sequences can be shown to be a non-contribution using qualitative arguments.	Anticipated Transient Without Scram (ATWS) does not consider the time of adverse moderator temperature coefficient (MTC). Loss of seal cooling, loss of all AC (SBO), inadvertent opening of power-operated relief valves (PORVs) and safety relief valves (SRVs) are	These issues have been addressed with the exception of the comment regarding throttling AFW after restoration of power following an SBO. Not modeling the operator action has an insignificant impact based on the two consequences of not throttling AFW, which

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		included in some, but not all event tree models. Assumption 8 in AS.1 states that operator action to throttle auxiliary feedwater (AFW) after power restoration following a SBO is assumed successful. No justification is provided for omitting this sequence.	are: • Premature draining of the Condensate Storage Tank, which is mitigated with offsite power available by supplying fire water to the suction of the AFW pumps. • Potential steam generator overfill, which could lead to failure of main steam line piping and therefore, loss of secondary heat removal capability. However, with offsite power available, once through cooling would be available to remove decay heat. The AFW throttle valves are potentially subject to the RITS 5b application. Therefore, if a change to the AFW throttle valve surveillance frequency is being evaluated as part of the RITS 5b process, a sensitivity study would be required to evaluate the impact of this gap.
SY-A19 [SY-A21] SY-B6 SY-B7	IDENTIFY system conditions that cause a loss of desired system function (e.g., excessive heat loads, excessive electrical loads, excessive humidity, etc.). PERFORM engineering analyses to determine the need for support systems that are plant-specific and reflect the variability in the conditions present during the postulated accidents for which the system is required to function. BASE support system modeling on realistic success criteria and timing, unless a	Room heatup calculations are planned to be performed as a part of the next MPS2 model update. However, that documentation does not appear to exist currently, or is not readily accessible. Also, no mention is made of electrical load shedding or excessive humidity conditions that could lead to a loss of function.	During the 2009 model update, the model and documentation were updated to address the supporting requirements not met. The failure of load shedding was added to the electric power fault tree. The accidents that result in excessive humidity such as steam line breaks (SLB) include failures of equipment where the SLB occurs. Room heatup calculations have been performed for the most risk significant rooms (i.e., switchgear rooms) and ventilation failures included in the model as appropriate. The ventilation systems components are

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	conservative approach can be justified (i.e., if their use does not impact risk significant contributors).		potentially subject to the RITS 5b application. Therefore, if a change to ventilation system components surveillance frequency is being evaluated as part of the RITS 5b process, a sensitivity study would be required to evaluate the impact of this gap.
SY-A20 [SY-A22]	TAKE CREDIT for system or component operability only if an analysis exists to demonstrate that rated or design capabilities are not exceeded.	The Component Cooling (CC) notebook mentions that a GOTHIC analysis was performed which stated that room cooling for the DC switchgear is needed only for equipment which requires DC power for more than one hour. However, the suggestion has been made that this analysis needs to be reviewed and other room heatup calculations need to be performed.	Room heatup calculations have been performed for the DC switchgear rooms and ventilation failures included in the model as appropriate. The ventilation systems components are potentially subject to the RITS 5b application. Therefore, if a change to ventilation system components surveillance frequency is being evaluated as part of the RITS 5b process, a sensitivity study would be required to evaluate the impact of this gap.
LE-A1 LE-C5 [LE-C6] LE-D6 [LE-D7]	IDENTIFY those physical characteristics at the time of core damage that can influence LERF. Examples include (a) RCS pressure (high RCS pressure can result in high pressure melt ejection) (b) status of emergency core coolant systems (failure in injection can result in a dry cavity and extensive Core Concrete Interaction) (c) status of containment isolation (failure of isolation can result in an unscrubbed release) (d) status of containment heat removal (e) containment integrity (e.g., vented, bypassed, or failed) (f) steam generator pressure and water level (PWRs) (g) status of containment inerting (BWRs). DEVELOP system models that support the accident progression analysis consistent	Include steam generator (SG) characteristics and containment isolation status in the Plant Damage State (PDS) binning, unless justification can be given for excluding them. SG characteristics are necessary for accurate induced steam generator tube rupture (SGTR) and SGTR initiating event LERF calculation, and containment isolation may be required if the valve closure has dependencies on other systems modeled in the Level 1 (e.g., isolation signal dependency on DC power and actuation logic). Include consideration of Emergency Core Cooling System (ECCS) / Low Pressure Safety Injection (LPSI) availability.	As part of the 2009 model update, the PDS tree was revised to specifically include availability of feedwater, which affects SG level. SG pressure is addressed in the Containment Event Tree (CET), which uses the NUREG-1570 methodology. This methodology bases the probability on the failure to close probability of an atmospheric dump valve (ADV). Since no support systems are required to close an ADV (i.e., they fail close on loss of air or power), there is no interaction with Level 1 and therefore it is appropriate to put it in the CET. However, containment isolation does require some support systems so it should be in the PDS tree using bridge trees. Per FSAR Table 5.2-11, all Containment

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	with the applicable requirements for Paragraph 4.5.4, as appropriate for the level of detail of the analysis. PERFORM containment isolation analysis in a realistic manner for the significant accident progression sequences resulting in a large early release. USE conservative or a combination of conservative or realistic treatment for the non-significant accident progression sequences resulting in a large early release. INCLUDE consideration of both the failure of containment isolation systems to perform properly and the status of safety systems that do not have automatic isolation provisions.		Isolation Valves (CIVs) that are not normally locked closed and are required to close post-accident are fail-closed valves. Therefore, the only support system required for success is the Engineered Safeguards Actuation System (ESAS), which produces the Containment Isolation Actuation Signal (CIAS). Penetrations with two active CIVs are treated as separate trains and therefore, receive train-specific CIAS signals. Consequently, for the containment isolation function to fail, both trains of CIAS would need to fail. As a result, this is considered an insignificant risk contributor. There are CIVs that open post-accident, which require support systems and operator action to close. The majority of the penetrations with these CIVs contain a check valve on the inside of containment, which require no operator action or support system to close. These penetrations are considered insignificant risk contributors. The only exceptions are the penetrations that contain the Containment Sump Isolation motor-operated valves (MOVs) as they do not contain an inside CIV. These MOVs open on a Sump Recirculation Actuation Signal (SRAS), which corresponds to low Refueling Water Storage Tank (RWST) level, to provide suction to the High Pressure Safety Injection (HPSI) and Containment Spray (CS) pumps during the sump recirculation phase. Failure of the open function is a significant core damage risk

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			contributor. In a core damage scenario, this penetration will either be full of water or closed (SRAS not generated) and therefore, does not represent a significant risk contributor. The CIVs are potentially subject to the RITS 5b application. Therefore, if a change to the CIV surveillance frequency is being evaluated as part of the RITS 5b process, a sensitivity study would be required to evaluate the impact of this gap.
LE-C2a [LE-C2] LE-C6 [LE-C7]	INCLUDE realistic treatment of feasible operator actions following the onset of core damage consistent with applicable procedures, e.g., Emergency Operating Procedures (EOPs)/Severe Accident Management Guidelines (SAMGs), proceduralized actions, or Technical Support Center guidance. In crediting Human Failure Events (HFEs) that support the accident progression analysis, USE the applicable requirements of Paragraph 4.5.5, as appropriate for the level of detail of the analysis.	IPE Table 4.8-3 is titled "Operator Action Basic Events", but no values for the actions are provided, and no detailed human error probabilities (HEP) calculation appears to have been performed. Per Table 4.8-4, a basic event probability of 0.1 was assigned to the probability of in-vessel recovery due to recovery of reactor pressure vessel (RPV) injection after core damage. No evaluation of the operator action is provided (the value was based on a value used in NUREG-4551). The SAMGs have not been reviewed for potential impact on the LERF, while certain actions could significantly affect the LERF. For example, opening RCS PORV prior to core damage can significantly reduce the chance of an induced SGTR.	Human Reliability Analysis (HRA) calculations were performed for the operator actions credited; ensuring dependencies with other operator actions are accounted for. The SAMGs have not yet been incorporated into the Level 2 model. However, the impact of not meeting this supporting requirement is that the current model is conservative. This gap has no impact on the RITS 5b application.
LE-C2b [LE-C3] LE-C8b [LE-C10]	REVIEW significant accident progression sequences resulting in a large early release to determine if repair of equipment can be credited. JUSTIFY credit given for repair [i.e., ensure that plant conditions do not preclude repair and actuarial data exists from which to estimate the repair failure probability (see SY-A22, DA-C14, and DA-	IPE Section 4.8.2 considers recovery events. It states that all recovery actions that involve AC power (HPSI, LPSI, CS, and Containment Air Recirculation (CAR) fan coolers) are accounted for in the Level 1 analysis. For other recoveries, Table 4.8.2-1 presents some recoveries, but the text indicates that these were treated as "house gates" that were set to zero. The CET used to	The sequences have not been reviewed for options available to reduce the LERF. However, the impact of not meeting this supporting requirement is that the current model is conservative. This gap has no impact on the RITS 5b application.

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	D8)]. AC power recovery based on generic data applicable to the plant is acceptable. REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions during accident progression that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences.	quantify the MPS2 Level 2 could not be found by Dominion, so the actual modeling could not be reviewed.	

Table 2 – Status of Gaps to NEI 00-02 and Capability Category II of ASME PRA Standard

Potential Impact of Gap on Implementation of RITS 5b				
Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
1	IE-A6 [IE-A8]	INTERVIEW plant personnel (e.g., Operations, Maintenance, Engineering, Safety Analysis) to determine if potential initiating events have been overlooked.	No documentation of plant personnel interviews to determine if potential initiating events have been overlooked was found in the PRA notebooks.	Documentation issue only, no impact on application.
2	IE-C1b [IE-C3]	CREDIT recovery actions [those implied in IE-C4(c), and those implied and discussed in IE-C6 through IE-C9] as appropriate. JUSTIFY each such credit (as evidenced such as through procedures or training.	Recovery actions are appropriately credited in the IE analysis and each such credit is justified (all credited actions are proceduralized). However, the SBO initiating event fault tree logic includes the potential to align to MPS3 power transformers or the SBO diesel. Such actions would occur after the SBO initiating event (available response times for the actions are approximately 100 minutes) and would appear to be more appropriately modeled in the post-initiator portions of the SBO logic (e.g., the power recovery function).	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.
3	AS-10	Dependencies among top events are identified and addressed.	Main Feedwater success criteria do not require makeup to the condenser when steam dump valves fail. No documentation of the verification that adequate volume exists in the condenser for successful cooldown. No modeling of makeup to the condenser was identified.	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.
4	AS-A4	For each modeled initiating event, using the success criteria defined for each key safety function (in accordance with SR SC-A4), IDENTIFY the necessary operator actions to achieve the defined success criteria.	In general, no summary or descriptions are provided for operator actions in either SC.1 or AS.1.	Documentation issue only, no impact on application.
5	AS-A7	DELINEATE the possible accident sequences for each modeled initiating event, unless the sequences can be shown to be a non-contribution using qualitative arguments.	ATWS does not consider the time of adverse MTC. Loss of seal cooling, loss of all AC (SBO), inadvertent opening of PORVs and SRVs are included in some, but not all event tree models. Assumption 8 in AS.1 states that operator action	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.

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Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
			to throttle AFW after power restoration following a SBO is assumed successful. No justification is provided for omitting this sequence.	
6	AS-A10	In constructing the accident sequence models, INCLUDE, for each modeled initiating event, sufficient detail that significant differences in requirements on systems and operator responses are captured. Where diverse systems and/or operator actions provide a similar function, if choosing one over another changes the requirements for operator intervention or the need for other systems, MODEL each separately.	While differences in system requirements for each initiating event may be included in the fault tree models, no delineation of how these differences impact operator actions or system responses is provided. For example, the success criteria for bleed and feed cooling are different between the General Plant Transient (GPT) and Main Feedwater (MFW) event models, however, no discussion is provided as to why.	Documentation issue only, no impact on application.
7	AS-B3	For each accident sequence, IDENTIFY the phenomenological conditions created by the accident progression. Phenomenological impacts include generation of harsh environments affecting temperature, pressure, debris, water levels, humidity, etc. that could impact the success of the system or function under consideration [e.g., loss of pump net positive suction head (NPSH), clogging of flow paths]. INCLUDE the impact of the accident progression phenomena, either in the accident sequence models or in the system models.	Only a limited discussion of phenomenological conditions created by the accident progression is provided in Section 2.3 of Volume AS.1. For example, the discussion provided on how a secondary line break outside containment affects the environmental conditions of equipment needed to mitigate the accident discusses the loss of IA, but no discussion is provided on the direct impact of a loss of MFW or any potential impact on AFW or the electrical switchgear rooms.	Documentation issue only, no impact on application. The IA compressors, 4160V and 480V switchgear rooms, and AFW system are located in the turbine building. Following a secondary line break outside containment, the IA compressors are expected to fail since they are not rated for a High Energy Line Break (HELB) environment. IA is a required support system for MFW; therefore, this dependency is directly accounted for in the system fault trees. The switchgear rooms are housed in Class I structures equipped with HELB doors and therefore, will

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				not be affected by a secondary line break. The AFW pumps and regulating valves are rated for a HELB environment and therefore, would not be affected by a secondary line break.
8	AS-B5a [AS-B6]	If plant configurations and maintenance practices create dependencies among various system alignments, DEFINE and MODEL these configurations and alignments in a manner that reflects these dependencies, either in the accident sequence models or in the system models.	The MPS2 model discusses how system configurations impact modeling in the system notebooks under the "Risk Monitor Considerations" section. However, no discussion is provided on how system alignments and configurations are applied when evaluating the PRA models outside of risk monitors.	Documentation issue only, no impact on application.
9	AS-C2	DOCUMENT the processes used to develop accident sequences and treat dependencies in accident sequences, including the inputs, methods, and results. For example, this documentation typically includes: (a) the linkage between the modeled initiating event in the Initiating Event Analysis section and the accident sequence model; (b) the success criteria established for each modeled initiating event including the bases for the criteria (i.e., the system capacities required to mitigate the accident and the necessary components required to achieve these capacities); (c) a description of the accident progression for each sequence or group of similar sequences (i.e., descriptions of the sequence timing, applicable procedural guidance, expected environmental or	A one-to-one correlation between each initiating event and the associated event tree is not clearly provided. The system success criteria and associated basis is not clearly provided. A discussion of the accident sequences will need to be revised pending resolution of issues associated with other AS supporting requirements. For example, the phenomenological conditions created by the accident. Operator actions needed are not clearly delineated along with any associated dependencies on system success or other operator actions (Refer to AS-B1, B3, and B6).	Documentation issue only, no impact on application.

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Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
		phenomenological impacts, dependencies between systems and operator actions, end states, and other pertinent information required to fully establish the sequence of events); (d) the operator actions reflected in the event trees, and the sequence-specific timing and dependencies that are traceable to the HRA for these actions; (e) the interface of the accident sequence models with plant damage states; (f) [when sequences are modeled using a single top event fault tree] the manner in which the requirements for accident sequence analysis have been satisfied.		
10	SC-B5	CHECK the reasonableness and acceptability of the results of the thermal/hydraulic, structural, or other supporting engineering bases used to support the success criteria. Examples of methods to achieve this include: (a) comparison with results of the same analyses performed for similar plants, accounting for differences in unique plant features (b) comparison with results of similar analyses performed with other plant-specific codes (c) check by other means appropriate to the particular analysis.	While the SC.1 and SC.2 make some comparisons to results from other plants (e.g., Calvert Cliffs Interim Reliability Evaluation Program (IREP)) for specific success criteria, there is no documented comparison of the overall set of MP2 success criteria to those of other plants. Also, as the Calvert Cliffs IREP has been superseded by more recent models, references to this older study may no longer be appropriate.	Documentation issue only, no impact on application.
11	SY-A4	PERFORM plant walkdowns and interviews with knowledgeable plant personnel (e.g., Engineering, Operations, etc.) to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	While the IPE documentation and conversations with the PRA engineers indicate that these tasks were performed, no documentation exists (walkdown sheets, system engineer interviews) to support this supposition.	Documentation issue only, no impact on application.

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Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
12	SY-A19 [SY-A21]	IDENTIFY system conditions that cause a loss of desired system function (e.g., excessive heat loads, excessive electrical loads, excessive humidity, etc.).	Room heatup calculations are planned to be performed as a part of the next MPS2 model update. However, that documentation does not appear to exist currently, or is not readily accessible. Also, no mention is made of electrical load shedding or excessive humidity conditions that could lead to a loss of function.	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.
13	SY-A20 [SY-A22]	TAKE CREDIT for system or component operability only if an analysis exists to demonstrate that rated or design capabilities are not exceeded.	The Component Cooling (CC) notebook mentions that a GOTHIC analysis was performed which stated that room cooling for the DC switchgear is needed only for equipment which requires DC power for more than one hour. However, the suggestion has been made that this analysis needs to be reviewed and other room heatup calculations need to be performed.	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.
14	SY-B6	PERFORM engineering analyses to determine the need for support systems that are plant-specific and reflect the variability in the conditions present during the postulated accidents for which the system is required to function.	As per SY-A19, room heatup calculations have not been performed. Systems that could fail based on excessive heat have not been properly documented.	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.
15	SY-B7	BASE support system modeling on realistic success criteria and timing, unless a conservative approach can be justified (i.e., if their use does not impact risk significant contributors).	As per SY-A19, room heatup calculations have not been performed. Systems that could fail based on excessive heat have not been properly documented.	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.
16	SY-B12 [SY-B11]	MODEL the ability of the available inventories of air, power, and cooling to support the mission time.	The system models for CC and IA do not appear to take credit for insufficient inventories. However, documentation of that appears insufficient.	Documentation issue only, no impact on application.
17	SY-C2	DOCUMENT the system functions and boundary, the associated success criteria,	No walkdown information, documentation of operating history, or room heatup calculations	Documentation issue only, no impact on application.

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Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
		the modeled components and failure modes including human actions, and a description of modeled dependencies including support system and common cause failures, including the inputs, methods, and results. For example, this documentation typically includes (a) system function and operation under normal and emergency operations (b) system model boundary (c) system schematic illustrating all equipment and components necessary for system operation (d) information and calculations to support equipment operability considerations and assumptions (e) actual operational history indicating any past problems in the system operation (f) system success criteria and relationship to accident sequence models (g) human actions necessary for operation of system (h) reference to system-related test and maintenance procedures (i) system dependencies and shared component interface (j) component spatial information (k) assumptions or simplifications made in development of the system models (l) the components and failure modes included in the model and justification for any exclusion of components and failure modes (m) a description of the modularization process (if used) (n) records of resolution of logic loops developed during fault tree linking (if used) (o) results of the system model evaluations (p) results of sensitivity studies (if used) (q)	exist.	

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		the sources of the above information (e.g., completed checklist from walkdowns, notes from discussions with plant personnel) (r) basic events in the system fault trees so that they are traceable to modules and to cutsets (s) the nomenclature used in the system models.		
18	DA-C10	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. COUNT only completed tests or unplanned operational demands as success for component operation. If the component failure mode is decomposed into sub-elements (or causes) that are fully tested, then USE tests that exercise specific sub-elements in their evaluation. Thus, one sub-element sometimes has many more successes than another. [Example: a diesel generator is tested more frequently than the load sequencer. IF the sequencer were to be included in the diesel generator boundary, the number of valid tests would be significantly decreased.]	There is no evidence in notebook DA.2 to indicate that a review of the surveillance procedures was performed.	Documentation issue only, no impact on application. During the 2009 model update, this data was obtained based on real plant data obtained from the station logs and the plant computer. Therefore, review of the procedures is not necessary.
19	DA-C15 [DA-C16]	Data on recovery from loss of offsite power, loss of service water, etc. are rare on a plant-specific basis. If available, for each recovery, COLLECT the associated recovery time with the recovery time being the period from identification of the system or function failure until the system or function is returned to service.	The DOM IE.2 notebook presents Offsite Power (OSP) frequencies with recovery presented in DOM HR.3 for all Dominion plants. OSP Recovery is calculated in DOM HR.3, but is not discussed (only presented in a spreadsheet). No specific assessment of the applicability of the events considered to the Millstone site is provided.	Documentation issue only, no impact on application. There were no plant specific LOOP events for MPS2 for the update period. Therefore, no plant-specific recovery times are available.

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20	IFPP-A3	For multi-unit sites with shared systems or structures, INCLUDE multi-unit areas, if applicable.	MPS2 is physically separate from MPS3 and shares no fluid systems or structures with MPS3. There is no potential for multi-unit flood scenarios; however, the documentation in notebook IF.1 should include discussion of why multi-unit flood areas (and scenarios) are not relevant for MPS2.	Documentation issue only, no impact on application.
21	IFSO-A5	For each source and its identified failure mechanism, IDENTIFY the characteristic of release and the capacity of the source. INCLUDE *(a)* a characterization of the breach, including type (e.g., leak, rupture, spray) *(b)* flow rate *(c)* capacity of source (e.g., gallons of water) *(d)* the pressure and temperature of the source.	The IF.1 and IF.2 notebooks consider leaks, ruptures and spray. The analysis generally considers sources of all sizes, which bounds the range of flow rates. The capacity of each source is considered qualitatively or quantitatively and potentially large sources are considered for their resulting impacts on the extent of flooding and propagation. Capacities of flood sources are also considered further in the IF.2 notebook. The documentation does not, however, discuss the pressures and temperatures of the sources. While most of the flood sources are relatively low temperature sources (e.g., service water, fire protection, etc.), high energy fluid sources are not highlighted, nor is there any discussion of whether the special characteristics of these sources might have unique plant effects.	Documentation issue only, no impact on application.
22	QU-B5	Fault tree linking and some other modeling approaches may result in circular logic that must be broken before the model is solved. BREAK the circular logic appropriately. Guidance for breaking logic loops is provided in NUREG/CR-2728 [Note (1)]. When resolving circular logic, AVOID introducing unnecessary conservatisms or non-conservatisms.	The MPS2 QU.1 and QU.2 notebooks do not include any discussion of the approach used for breaking circular logic loops. (The discussion in QU.1 Attachment 1 on Revision 4 does mention that changes were made to system fault trees to correct circular logic related to consequential Loss of Coolant Accidents (LOCAs), and Section 2.2.1 of QU.2 notes that logic loops related to DC ventilation changes were	Documentation issue only, no impact on application.

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			addressed. Finally, Tables 17-19 in QU.2 identify changes in model results due to removal of logic loops.) Table 1 of the systems analysis assumptions notebook (SY.2) includes several specific entries regarding the creation of circular logic cut gates to break logic loops in the AC, DC, Engineered Safety Feature Actuation System (ESFAS), Heating, Ventilation and Air Conditioning (HVAC), IA, and Service Water systems, but there is no discussion on where/how these gates break the logic. Instead, these assumptions and the response to comment 7 in Attachment 2 to SY.2 refer to documentation on the necessity of these gates in the final quantification documentation (QU.2), but no such documentation exists.	
23	QU-E3	ESTIMATE the uncertainty interval of the overall CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G9, IE-C13), taking into account the "state-of-knowledge" correlation.	No parametric uncertainty analysis has been performed for the MPS2 PRA.	Documentation issue only, no impact on application.
24	QU-E4	EVALUATE the sensitivity of the results to key model uncertainties and key assumptions using sensitivity analyses.	No evaluation of the model uncertainties and assumptions have been performed or documented.	Documentation issue only, no impact on application.
25	LE-A1	IDENTIFY those physical characteristics at the time of core damage that can influence LERF. Examples include (a) RCS pressure (high RCS pressure can result in high pressure melt ejection) (b) status of emergency core coolant systems (failure in injection can result in a dry cavity and extensive Core Concrete Interaction) (c)	Include SG characteristics and containment isolation status in the PDS binning, unless justification can be given for excluding them. SG characteristics are necessary for accurate induced SGTR and SGTR initiating event LERF calculation, and containment isolation may be required if the valve closure has dependencies on other systems modeled in the Level 1 (e.g.,	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.

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Potential Impact of Gap on Implementation of RITS 5b				
Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
		status of containment isolation (failure of isolation can result in an unscrubbed release) (d) status of containment heat removal (e) containment integrity (e.g., vented, bypassed, or failed) (f) steam generator pressure and water level (PWRs) (g) status of containment inerting (BWRs).	isolation signal dependency on DC power and actuation logic). Include consideration of ECCS/LPSI availability.	
26	LE-C2a [LE-C2]	INCLUDE realistic treatment of feasible operator actions following the onset of core damage consistent with applicable procedures, e.g., EOPs/SAMGs, proceduralized actions, or Technical Support Center guidance.	IPE Table 4.8-3 is titled "Operator Action Basic Events", but no values for the actions are provided, and no detailed HEP calculation appears to have been performed. Per Table 4.8-4, a basic event probability of 0.1 was assigned to the probability of in-vessel recovery due to recovery of RPV injection after core damage. No evaluation of the operator action is provided (the value was based on a value used in NUREG-4551). The SAMGs have not been reviewed for potential impact on the LERF, while certain actions could significantly affect the LERF. For example, opening RCS PORV prior to core damage can significantly reduce the chance of an induced SGTR.	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.
27	LE-C2b [LE-C3]	REVIEW significant accident progression sequences resulting in a large early release to determine if repair of equipment can be credited. JUSTIFY credit given for repair [i.e., ensure that plant conditions do not preclude repair and actuarial data exists from which to estimate the repair failure probability (see SY-A22, DA-C14, and DA-D8)]. AC power recovery based on generic data applicable to the plant is acceptable.	IPE Section 4.8.2 considers recovery events. It states that all recovery actions that involve AC power (HPSI, LPSI, CAR fan coolers and containment sprays) are accounted for in the Level 1 analysis. For other recoveries, Table 4.8.2-1 presents some recoveries, but the text indicates that these were treated as "house gates" that were set to zero. The CET used to quantify the MPS2 Level 2 could not be found by Dominion, so the actual modeling could not be reviewed.	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.

Table 2 – Status of Gaps to NEI 00-02 and Capability Category II of ASME PRA Standard

Potential Impact of Gap on Implementation of RITS 5b				
Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
28	LE-C5 [LE-C6]	DEVELOP system models that support the accident progression analysis consistent with the applicable requirements for paragraph 4.5.4, as appropriate for the level of detail of the analysis.	System models that affect the accident progression (e.g., sprays and containment heat removal) were developed and documented in the applicable system analysis notebooks (SY). However, a containment isolation fault tree may also be required, as it appears from IPE Section 4.4.5 that the isolation valves may require modeling of dependencies. The containment isolation document (2-PRA-93-032) could not be located by Dominion, so the actual system requirements are not clear.	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.
29	LE-C6 [LE-C7]	In crediting HFEs that support the accident progression analysis, USE the applicable requirements of paragraph 4.5.5, as appropriate for the level of detail of the analysis.	System level operator actions are described in the Level 1 System Analysis notebooks. Offsite power recovery probabilities are maintained within the Level 1 Data Analysis. SAMGs have not been incorporated into the MPS2 Level 2 analysis, although credit for initiation of low pressure injection after the onset of core damage was combined with hardware failures, and assigned a total probability of 0.1. IPE Table 4.8-3 (page 4-149) shows three other operator action basic events in the Level 2, although no HEP was presented.	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.
30	LE-C8a [LE-C9]	JUSTIFY any credit given for equipment survivability or human actions under adverse environments.	It appears that some consideration was given, as seen on pages 4-140 (#29) and F-10 of the IPE, which state consideration of containment sprays being failed by the accident progression. Page F-21 shows a probability of 1E-2 that sprays are failed by the accident progression, although the only basis is an assumption on page 4-162. Other than the sprays, it does not appear that any other equipment survivability was examined, except that no credit was given	Documentation issue only, no impact on application.

Table 2 – Status of Gaps to NEI 00-02 and Capability Category II of ASME PRA Standard

Potential Impact of Gap on Implementation of RITS 5b				
Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
			to operation after containment failure. The equipment survivability should be examined and explicitly discussed to meet the supporting requirement. In general, equipment is capable (and credited) of performing at levels significantly worse than the design basis conditions. For example, even though spray headers and SG equipment are credited up until containment failure (pressures and temperatures far greater than design basis), they will be subject to worse than design basis conditions in a severe accident. Such credit should be provided in the documentation.	
31	LE-C8b [LE-C10]	REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions during accident progression that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences.	The significant accident progression sequences were not reviewed explicitly for the consideration of continued equipment operation or operator actions to reduce the LERF.	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.
32	LE-C9b [LE-C12]	REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions after containment failure that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences.	Although a review of the significant accident progression sequences for post-containment failure operation might not identify any potential for LERF reduction, the review should be performed and documented to meet the supporting requirement.	Documentation issue only, no impact on application.

Table 2 – Status of Gaps to NEI 00-02 and Capability Category II of ASME PRA Standard

Potential Impact of Gap on Implementation of RITS 5b				
Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
33	LE-D1b [LE-D2]	EVALUATE the impact of accident progression conditions on containment seals, penetrations, hatches, drywell heads (BWRs), and vent pipe bellows. INCLUDE these impacts as potential containment challenges, is required. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	IPE Section 4.4.2 notes the consideration of penetrations, hatch failure, etc. The complete report is documented in the 1993 EQE Engineering calculation 52204-R-002, as referenced in Section 4.4.1 of the IPE. However, the EQE report was not available for review, and should be brought into the Dominion document control.	Documentation issue only, no impact on application.
34	LE-D6 [LE-D7]	PERFORM containment isolation analysis in a realistic manner for the significant accident progression sequences resulting in a large early release. USE conservative or a combination of conservative or realistic treatment for the non-significant accident progression sequences resulting in a large early release. INCLUDE consideration of both the failure of containment isolation systems to perform properly and the status of safety systems that do not have automatic isolation provisions.	Containment isolation was discussed in IPE Section 4.4.5, which references a detailed evaluation in MPS2 calculation 2-PRA-93-032 (July 1993). However, this calculation could not be located by Dominion, so the details could not be reviewed. The IPE states that isolation failure is dominated by a 2" line (failure of three air-operated valves (AOVs)) and two 6" hydrogen purge lines (two AOVs each). The analysis needs to consider if the AOVs require an actuation signal; if so, then their fault tree solutions should be tied to the sequence logic to capture dependencies. The analysis needs to provide a basis for small vs. large containment isolation failures. Also, there are two references in the IPE citing "personal communication" with individuals. References to memoranda or something similar should be provided. Section 2.4 of the AS.1 notebook states that "Since the Containment is operated at sub-atmospheric pressure the probability of Containment bypass as a result of failure to isolate is very low for all sequences. Hence this function has been excluded from individual event trees." This	Potential logic model issue. The impact of not meeting this element on the RITS 5b application is required to be reviewed.

Table 2 – Status of Gaps to NEI 00-02 and Capability Category II of ASME PRA Standard

Potential Impact of Gap on Implementation of RITS 5b				
Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
			apparently is an attempt to justify the isolation failure not being linked to the other functions, but a quantitative evaluation would be required to justify such a statement.	
35	LE-F1a [LE-F1]	PERFORM a quantitative evaluation of the relative contribution to LERF from plant damage states and significant LERF contributors from Table 4.5.9-3.	QU.2 Section 2.3.2 provided LERF by initiating event, Section 2.3.6 presented the dominant LERF cutsets, Section 2.3.9 presents the LERF importance analysis, and Table 15 presents the system contribution to LERF. PRA00YQA-03015S2, Rev. 1 quantified the LERF by PDS, but the evaluation was in 2001 and the PDS quantification has not been documented for the current results. "Significant LERF contributors" have not been defined in QU.2.	Documentation issue only, no impact on application.
36	LE-F1b [LE-F2]	REVIEW contributors for reasonableness (e.g., to assure excessive conservatisms have not skewed the results, level of plant specificity is appropriate for significant contributors, etc.).	Section 2.4.11 of QU.2 examines some potential plant improvements to reduce the CDF, but does not select potential improvements based on the dominant LERF contributors. Section 2.3.6 presented the dominant LERF cutsets, but did not discuss their potential for excess conservatism.	Documentation issue only, no impact on application.
37	LE-G2	DOCUMENT the process used to identify plant damage states and accident progression contributors, define accident progression sequences, evaluate accident progression analyses of containment capability, and quantify and review the LERF results. For example, this documentation typically includes (a) the plant damage states and their attributes, as used in the analysis (b) the method used to bin the accident sequences into plant damage states (c) the containment failure	The PDS documentation was created in the IPE and has not been updated even though there have been many updates to the Level 1 analysis. The IPE PDS binning documentation does not provide sufficient detail about specific sequence binning. The CET is documented in the IPE, but it is difficult to follow the exact logic or even the exact values used for split fraction basic events for each PDS.	Documentation issue only, no impact on application.

Table 2 – Status of Gaps to NEI 00-02 and Capability Category II of ASME PRA Standard

Potential Impact of Gap on Implementation of RITS 5b				
Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
		modes, phenomena, equipment failures and human actions considered in the development of the accident progression sequences and the justification for their inclusion or exclusion from the accident progression analysis (d) the treatment of factors influencing containment challenges and containment capability, as appropriate for the level of detail of the analysis (e) the basis for the containment capacity analysis including the identification of containment failure location(s), if applicable (f) the accident progression analysis sequences considered in the containment event trees (g) the basis for parameter estimates (h) the model integration process including the results of the quantification including uncertainty and sensitivity analyses, as appropriate for the level of detail of the analysis.		
38	LE-G3	DOCUMENT the relative contribution of contributors (i.e., plant damage states, accident progression sequences, phenomena, containment challenges, containment failure modes) to LERF.	The PDS contribution was tallied in PRA00YQA-03015S2 in 1991, but has not been updated in the current QU or LE notebooks. The QU notebooks tabulate LERF by initiating event and system contribution, but not by the contribution due to various phenomena or containment challenges.	Documentation issue only, no impact on application.
39	LE-G4	DOCUMENT key assumptions and key sources of uncertainty associated with the LERF analysis, including results and important insights from sensitivity studies.	The IPE Section 4.2.2 presents a list of sensitivities to be evaluated by the Modular Accident Analysis Program (MAAP) code, but does not actually discuss their evaluation. However, many sensitivities are mentioned in various subsections, but it would be helpful to	Documentation issue only, no impact on application.

Table 2 – Status of Gaps to NEI 00-02 and Capability Category II of ASME PRA Standard

Potential Impact of Gap on Implementation of RITS 5b				
Gap Number	**Element Not Met**	**Element Description**	**Review Comment**	**Impact on RITS 5b Application**
			compile a list of the sensitivities performed and present their conclusions. The QU.4 document does a good job of identifying key sources of uncertainty, but does not identify the specific assumptions from the IPE. In the IPE, the assumptions were stated as they were used, but were not tabulated and only a few were selected for sensitivity analysis. QU.4 Table 10 documents sensitivities that vary the HEPs and CCF probabilities, and Table 11 identifies some sensitivities based on Level 1 assumptions. However, per Table 12, the sensitivities have not been completed, and in any case, no sensitivities were identified based on the Level 2 analysis. The sensitivity analyses should be expanded and should be performed on the updated models.	
40	LE-G5	IDENTIFY limitations in the LERF analysis that would impact applications.	Section 2.4.12 of the QU.2 notebook states that the QU.4 notebook will identify model limitations, but they are not identified in QU.4.	Documentation issue only, no impact on application.

3.6 External Events Considerations

The NEI 04-10, Revision 1 methodology allows for STI change evaluations to be performed in the absence of quantifiable PRA models for all external hazards. For those cases where the STI cannot be modeled in the plant PRA (or where a particular PRA model does not exist for a given hazard group), a qualitative or bounding analysis is performed to provide justification for the acceptability of the proposed test interval change.

The external event considerations were derived from the MPS2 Individual Plant Examination - External Events (IPEEE) (Ref. 6.11). For events such as fire, seismic, extreme winds and other external events, the risk assessments from the IPEEE can be used for insights on changes to surveillance intervals.

Fire Risk

The MPS2 PRA does not include a fire model. Therefore, the results of the fire risk assessment performed for the IPEEE can be qualitatively assessed for insights on changes to surveillance intervals. The IPEEE fire risk analysis quantified a core damage frequency (CDF) by using a combination of Fire Induced Vulnerability Evaluation (FIVE) methodology and Fire PRA. The CDF due to fires is 6.3E-06/yr, with the dominant risk being fires in the auxiliary building, turbine building, cable vault, and intake structure.

Seismic Risk

The MPS2 PRA does not include a seismic model. Therefore, the results of the seismic risk assessment performed for the IPEEE can be qualitatively assessed for insights on changes to surveillance intervals. The IPEEE seismic risk analysis used the EPRI Seismic Margins Method to determine seismic vulnerabilities beyond design basis and therefore, did not calculate a seismic CDF. This process utilized a screening process to identify components that are considered not seismically rugged and required further evaluation. STI changes associated with these components would require investigation within the RITS 5b process.

High Winds, Floods and Other External Events

The risk of other external events such as high winds, external floods, transportation accidents, and weather-related events were assessed in the MPS2 IPEEE. This process utilized a screening process to identify components that required further evaluation. STI changes associated with these components would require investigation within the RITS 5b process.

3.7 Summary

The MPS2 PRA technical capability evaluations and the maintenance and update processes described above provide a robust basis for concluding that the full power internal events MPS2 PRA is suitable for use in risk-informed processes such as that proposed for the implementation of a SFCP. In performing the assessments for the other hazard groups, the qualitative or bounding approach will be utilized in most cases. Also, in addition to the standard set of sensitivity studies required per the NEI 04-10, Revision 1 methodology, open items for changes at the site and remaining gaps to specific requirements in the 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 RESULTS

None

5.0 CONCLUSIONS

The MPS2 PRA model supports the RITS 5b application.

6.0 REFERENCES

6.1. CE NPSD-1182-P, *Millstone Nuclear Station Unit 2 Probabilistic Safety Assessment Peer Review Report*, Final Report, Task 1037, Combustion Engineering Owners Group, January 2000
6.2. MPS2 Probabilistic Risk Assessment Model Notebook Part IV Support Information, *Appendix A – PRA Model Reviews*, Revision 2, May 2011
6.3. MPS2 Probabilistic Risk Assessment Model Notebook Part IV, *Appendix A.1, Internal Events Model Self Assessment*, Revision 2, February 2011
6.4. Peach Bottom Atomic Power Station, *Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)*, ADAMS ML092470153, August 31, 2009
6.5. *Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies*, Industry Guidance Document, NEI 04-10, Revision 1, April 2007
6.6. ASME RA-S-2002, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*, with ASME RA-Sa-2003 and RA-Sb-2005 Addenda, ASME, 2005
6.7. US Nuclear Regulatory Commission, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Regulatory Guide 1.200, Revision 1, January 2007
6.8. MPS2 2009 PRA Model, *External Release of MPS2 PRA Model M209Aa*, MEMO-PRA-20110009, Revision 0, August 25, 2011
6.9. MPS2 Probabilistic Risk Assessment Quality Summary Notebook Part IV Support Information, *Appendix B – Quality Summary*, Revision 0, May 2011

6.11. Millstone Unit 2 Nuclear Power Plant, *Individual Plant Examination of External Events*, Summary Report, December 29, 1995

6.12. ASME/ANS RA-S-2008, *Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications* and its 2009 addendum (ASME/ANS RA-Sa-2009)

6.13. North Anna Power Station, CO-NRC-000-10-122, *Virginia Electric and Power Company (Dominion) North Anna Power Station Units 1 and 2 Proposed License Amendment Request Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)*, March 30, 2010

6.14. Surry Power Station, CO-NRC-000-10-183, *Virginia Electric and Power Company (Dominion) Surry Power Station Units 1 and 2 Proposed License Amendment Request Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)*, March 30, 2010

6.15. Millstone Power Station Unit 3, Adams Accession Number ML11193A225, *Dominion Nuclear Connecticut, Inc. Millstone Power Station Unit 3 License Amendment Request to Relocate TS Surveillance Frequencies to Licensee Controlled Program in Accordance with TSTF-425, Revision 3*, July 5, 2011

ATTACHMENT 3

Marked-up Technical Specifications Changes

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

SECTION PAGE

1.0 DEFINITIONS

INDEX

ADMINISTRATIVE CONTROLS

AZIMUTHAL POWER TILT - T_q

1.18 AZIMUTHAL POWER TILT shall be the difference between the maximum power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

$$\text{AZIMUTHALPOWERTILT} = \left[\frac{\text{Maximum power in any core quadrant (upper or lower)}}{\text{Average power of all quadrants (upper or lower)}}\right] - 1$$

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (micro-curie/gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

DOSE EQUIVALENT XE-133

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

~~STAGGERED TEST BASIS~~ ← DELETED

~~1.21 A STAGGERED TEST BASIS shall consist of:~~

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

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

FREQUENCY NOTATION

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

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 6 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Prior to each release.
N.A.	Not applicable.
SFCP	At the frequency specified in the Surveillance Frequency Control Program.

~~September 25, 2003~~

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - (SDM)

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 3(1)*, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN not within the limit specified in the CORE OPERATING LIMITS REPORT, within 15 minutes, initiate and continue boration at ≥ 40 gpm of boric acid solution at or greater than the required refueling water storage tank (RWST) concentration (ppm) until the SHUTDOWN MARGIN is restored to within limit.

SURVEILLANCE REQUIREMENTS

4.1.1.1 Verify SHUTDOWN MARGIN is within the limit specified in the CORE OPERATING LIMITS REPORT at ~~least once every 24 hours~~.

the frequency specified in the Surveillance Frequency Control Program

*(1) See Special Test Exception 3.10.1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 REACTIVITY CONTROL SYSTEMS

REACTIVITY BALANCE

LIMITING CONDITION FOR OPERATION

3.1.1.2 The core reactivity balance shall be within ± 1% Δk/k of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTION:

With core reactivity balance not within limit:

Re-evaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation and establish appropriate operating restrictions and Surveillance Requirements within 7 days or otherwise be in MODE 3 within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.2 Verify*(1) overall core reactivity balance is within ± 1% Δk/k of predicted values prior to entering MODE 1 after fuel loading and at ~~least once every 31 Effective Full Power Days~~ **(2). The provisions of Specification 4.0.4 are not applicable.

the frequency specified in the Surveillance Frequency Control Program

*(1) The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

**(2) Only required after 60 Effective Full Power Days.

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be ≥ 515°F when the reactor is critical.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the Reactor Coolant System temperature (T_{avg}) < 515°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be ≥ 515°F.

a. Within 15 minutes prior to making the reactor critical, and

b. At ~~least once per hour~~ when the reactor is critical and the Reactor Coolant System temperature (T_{avg}) is < 525°F.

the frequency specified in the Surveillance Frequency Control Program

* With K_{eff} ≥ 1.0.

ACTION: (Continued):

C. CEA Deviation Circuit inoperable.	C.1 Verify the indicated position of each CEA to be within 10 steps of all other CEAs in its group within 1 hour and every 4 hours thereafter or otherwise be in MODE 3 within the next 6 hours.
D. One or more CEAs untrippable. OR Two or more CEAs misaligned by ≥ 20 steps.	D.1 Be in MODE 3 within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 Verify the indicated position of each CEA to be within 10 steps of all other CEAs in its group at ~~least once per 12 hours~~ AND within 1 hour following any CEA movement larger than 10 steps.

4.1.3.1.2 Verify CEA freedom of movement (trippability) by moving each individual CEA that is not fully inserted into the reactor core 10 steps in either direction at ~~least once per 92 days~~.

4.1.3.1.3 Verify the CEA Deviation Circuit is OPERABLE at ~~least once per 92 days~~ by a functional test of the CEA group Deviation Circuit which verifies that the circuit prevents any CEA from being misaligned from all other CEAs in its group by more than 10 steps (indicated position).

4.1.3.1.4 Verify the CEA Motion Inhibit is OPERABLE by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit prevents regulating CEAs from being inserted beyond the Transient Insertion Limits specified in the CORE OPERATING LIMITS REPORT:

a. Prior to each entry into MODE 2 from MODE 3, except that such verification need not be performed more often than once per 31 days, and

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

the frequency specified in the Surveillance Frequency Control Program

LIMITING CONDITION FOR OPERATION (Continued)

b) The CEA group(s) with the inoperable indicator is fully inserted, and subsequently maintained fully inserted, while maintaining the withdrawal sequence and THERMAL POWER level required by Specification 3.1.3.6 and when this CEA group reaches its fully inserted position, the "Full In" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6.

4. If the failure of the position indicator channel(s) is during STARTUP, the CEA group(s) with the inoperable position indicator channel must be moved to the "Full Out" position and verified to be fully withdrawn via a "Full Out" indicator within 4 hours.

c. With a maximum of one reed switch position indicator channel per group or one pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel at either its fully inserted position or fully withdrawn position, operation may continue provided:

1. The position of this CEA is verified immediately and at least once per 12 hours thereafter by its "Full In" or "Full Out" limit (as applicable).

2. The fully inserted CEA group(s) containing the inoperable position channel is subsequently maintained fully inserted, and

3. Subsequent operation is within the limits of Specification 3.1.3.6.

d. With one or more pulse counting position indicator channels inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided all of the reed switch position indicator channels are OPERABLE.

SURVEILLANCE REQUIREMENTS

required

4.1.3.3 Each position indicator channel shall be determined to be OPERABLE by verifying the pulse counting position indicator channels and the reed switch position indicator channels agree within 6 steps at ~~least once per 12 hours~~.

the frequency specified in the Surveillance Frequency Control Program

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual CEA drop time, from a fully withdrawn position, shall be $\leq$2.75 seconds from when electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515°$ F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time shall be demonstrated through measurement with $T_{avg} \geq 515°$ F, and all reactor coolant pumps operating prior to reactor criticality:

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

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

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

the frequency specified in the Surveillance Frequency Control Program

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to ≥ 176 steps.

APPLICABILITY: MODE 1*(1)
MODE 2(1),(2)** with any regulating CEA not fully inserted.

ACTION:

INOPERABLE EQUIPMENT	REQUIRED ACTION
A. One or more shutdown CEAs not within limit.	A.1 Restore shutdown CEA(s) to within limit within 2 hours or otherwise be in MODE 3 within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Verify each shutdown CEA is withdrawan ≥ 176 steps at ~~least once per 12 hours~~.

the frequency specified in the Surveillance Frequency Control Program

*(1) This LCO is not applicable while performing Specification 4.1.3.1.2.

**(2)See Special Test Exceptions 3.10.1 and 3.10.2.

REGULATING CEA INSERTION LIMITS (Continued)

B. Regulating CEA groups inserted between the Long Term Steady State Insertion limit and the Transient Insertion Limit specified in the CORE OPERATING LIMITS REPORT for intervals > 4 hours per 24 hour interval.	B.1 Verify Short Term Steady State Insertion Limits as specified in the CORE OPERATING LIMITS REPORT are not exceeded within 15 minutes or otherwise be in MODE 3 within the next 6 hours. OR B.2 Restrict increases in THERMAL POWER to < 5% RATED THERMAL POWER per hour within 15 minutes or otherwise be in MODE 3 within the next 6 hours.
C. Regulating CEA groups inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit specified in the CORE OPERATING LIMITS REPORT for intervals > 5 effective full power days (EFPD) per 30 EFPD or interval > 14 EFPD per 365 EFPD.	C.1 Restore regulating CEA groups to within the Long Term Steady State Insertion Limit specified in the CORE OPERATING LIMITS REPORT within 2 hours or otherwise be in MODE 3 within the next 6 hours.
D. PDIL alarm circuit inoperable.	D.1 Perform Specification 4.1.3.6.1 within 1 hour and once per 4 hours thereafter or otherwise be in MODE 3 within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6.1 Verify each regulating CEA group position is within the Transient Insertion Limits specified in the CORE OPERATING LIMITS REPORT at ~~least once per 12 hours~~. The provisions of Specification 4.0.4 are not applicable for entering into MODE 2 from MODE 3.

4.1.3.6.2 Verify the accumulated times during which the regulating CEA groups are inserted beyond the Steady State Insertion Limits but within the Transient Insertion Limits specified in the CORE OPERATING LIMITS REPORT at ~~least once per 24 hours~~.

4.1.3.6.3 Verify PDIL alarm circuit is OPERABLE at ~~least once per 31 days~~.

the frequency specified in the Surveillance Frequency Control Program

CONTROL ROD DRIVE MECHANISMS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod drive mechanisms shall be de-energized.

APPLICABILITY: MODES 3*, 4, 5 and 6, whenever the RCS boron concentration is less than refueling concentration of Specification 3.9.1.

ACTION:

With any of the control rod drive mechanisms energized, restore the mechanisms to their de-energized state within 2 hours or immediately open the reactor trip circuit breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The control rod drive mechanisms shall be verified to be de-energized at ~~least once per 24 hours~~.

the frequency specified in the Surveillance Frequency Control Program

* The control rod drive mechanisms may be energized for MODE 3 as long as 4 reactor coolant pumps are OPERATING, the reactor coolant system temperature is greater than 500° F, the pressurizer pressure is greater than 2000 psia and the high power trip is OPERABLE.

SURVEILLANCE REQUIREMENTS (Continued)

the frequency specified in the Surveillance Frequency Control Program

4.2.1.2 Excore Detector Monitoring System*(1) - The excore detector monitoring system may be used for monitoring the core power distribution ~~by~~:

a. Verifying at ~~least once per 12 hours~~ that the CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6.

b. Verifying at ~~least once per 31 days~~ that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the allowable limits specified in the CORE OPERATING LIMITS REPORT.

4.2.1.3 Incore Detector Monitoring System**(2), ***(3) - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated ~~at least once per 31 days~~.

b. Have their alarm setpoint adjusted to less than or equal to the limits specified in the CORE OPERATING LIMITS REPORT.

at the frequency specified in the Surveillance Frequency Control Program,

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~~*~~(1) Only required to be met when the Excore Detector Monitoring System is being used to determine Linear Heat Rate.

**(2) Only required to be met when the Incore Detector Monitoring System is being used to determine Linear Heat Rate.

***(3) Not required to be performed below 20% RATED THERMAL POWER.

TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F^T_r

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F^T_r shall be within the 100% power limit specified in the CORE OPERATING LIMITS REPORT. The F^T_r value shall include the effect of AZIMUTHAL POWER TILT.

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

ACTION:

With F^T_r exceeding the 100% power limit within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_r to within the power dependent limit specified in the CORE OPERATING LIMITS REPORT and withdraw the CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F^T_r shall be determined to be within the 100% power limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At ~~least once per 31 days of accumulated operation~~ in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.020.

4.2.3.3 F^T_r shall be determined by using the incore detectors to obtain a power distribution map with all CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump Combination.

the frequency specified in the Surveillance Frequency Control Program

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* See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

AZIMUTHAL POWER TILT - T_Q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall be ≤ 0.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% of RATED THERMAL POWER[(1)]*.

ACTION:

a. With the indicated $T_q > 0.02$ but ≤ 0.10, either restore T_q to ≤ 0.02 within 2 hours or verify the TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR (F^T_r) is within the limit of Specification 3.2.3 within 2 hours and once per 8 hours thereafter. Or otherwise, reduce THERMAL POWER to $\leq$ 50% of RATED THERMAL POWER within the next 4 hours.

b. With the indicated $T_q > 0.10$, perform the following actions: [(2)]**

1. Verify the TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR (F^T_r) is within the limit of Specification 3.2.3 within 2 hours; and

2. Reduce THERMAL POWER to $\leq$ 50% of RATED THERMAL POWER within 2 hours; and

3. Restore $T_q \leq 0.02$ prior to increasing THERMAL POWER. 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 measured T_q is verified ≤ 0.02 at least once per hour for 12 hours, or until verified at 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4.1 Verify T_q is within limit at ~~least once every 12 hours~~. The provisions of Specification 4.0.4 are not applicable for entering into MODE 1 with THERMAL POWER > 50% of RATED THERMAL POWER from MODE 1.

the frequency specified in the Surveillance Frequency Control Program

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*[(1)]See Special Test Exception 3.10.2.

**[(2)]All subsequent Required ACTIONS must be completed if power reduction commences prior to restoring $T_q \leq 0.10$.

DNB MARGIN

LIMITING CONDITION FOR OPERATION

3.2.6 The DNB margin shall be preserved by maintaining the cold leg temperature, pressurizer pressure, reactor coolant flow rate, and AXIAL SHAPE INDEX within the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.6.1 The cold leg temperature, pressurizer pressure, and AXIAL SHAPE INDEX shall be determined to be within the limits specified in the CORE OPERATING LIMITS REPORT at ~~least once per 12 hours~~. The reactor coolant flow rate shall be determined to be within the limit specified in the CORE OPERATING LIMITS REPORT at ~~least once per 31 days~~.

4.2.6.2 The provisions of Specification 4.0.4 are not applicable.

the frequency specified in the Surveillance Frequency Control Program

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

required

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

4.3.1.1.2 The bypass function and automatic bypass removal function shall be demonstrated OPERABLE during a CHANNEL FUNCTIONAL TEST once within 92 days prior to each reactor startup. The total bypass function shall be demonstrated OPERABLE at ~~least once per 18 months~~ during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

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

the frequency specified in the Surveillance Frequency Control Program

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

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.	Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2.	Power Level - High				
	a. Nuclear Power	~~S~~	~~D~~(2), ~~M~~(3), ~~Q~~(5)	~~M~~	1, 2, 3*
	b. ΔT Power	~~S~~	~~D~~(4), ~~Q~~	~~M~~	1
3.	~~Reactor~~ Coolant Flow - Low	~~S~~	~~R~~	~~M~~	1, 2
4.	Pressurizer Pressure - High	~~S~~	~~R~~	~~M~~	1, 2
5.	Containment Pressure - High	~~S~~	~~R~~	~~M~~	1, 2
6.	Steam Generator Pressure - Low	~~S~~	~~R~~	~~M~~	1, 2
7.	Steam Generator Water Level - Low	~~S~~	~~R~~	~~M~~	1, 2
8.	Local Power Density - High	~~S~~	~~R~~	~~M~~	1
9.	Thermal Margin/Low Pressure	~~S~~	~~R~~	~~M~~	1, 2
10.	Loss of Turbine--Hydraulic Fluid Pressure - Low	N.A.	~~R~~	S/U(1)	N.A.

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

TABLE 4.3-1 ~~(Continued)~~

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
11.	Wide Range Logarithmic Neutron Flux Monitor - Shutdown	~~S~~	~~R~~(5)	S/U(1)	3, 4, 5
12.	DELETED				
13.	Reactor Protection System Logic Matrices	N.A.	N.A.	~~M~~ and S/U(1)	1, 2 and *
14.	Reactor Protection System Logic Matrix Relays	N.A.	N.A.	~~M~~ and S/U(1)	1, 2 and ~~*~~
15.	Reactor Trip Breakers	N.A.	N.A.	~~M~~	1, 2 and *

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The engineered safety feature actuation system instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an engineered safety feature actuation system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, either adjust the trip setpoint to be consistent with the value specified in the Trip Setpoint column of Table 3.3-4 within 2 hours or declare the channel inoperable and take the ACTION shown in Table 3.3-3.

b. With an engineered safety feature actuation system instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

required

4.3.2.1.1 Each engineered safety feature actuation system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The bypass function and automatic bypass removal function shall be demonstrated OPERABLE during a CHANNEL FUNCTIONAL TEST once within 92 days prior to each reactor startup. The total bypass function shall be demonstrated OPERABLE at ~~least once per 18 months~~ during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

the frequency specified in the Surveillance Frequency Control Program

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

the frequency specified in the Surveillance Frequency Control Program

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

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	~~R~~	N. A.
b. Containment Pressure - High	~~S~~	~~R~~	~~M~~	1, 2, 3
c. Pressurizer Pressure - Low	~~S~~	~~R~~	~~M~~	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	~~M~~(1)	1, 2, 3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	~~R~~	N.A.
b. Containment Pressure-- High - High	~~S~~	~~R~~	~~M~~	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	~~M~~(1)	1, 2, 3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	~~R~~	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	~~R~~	N.A.
c. Containment Pressure - High	~~S~~	~~R~~	~~M~~	1, 2, 3
d. Pressurizer Pressure - Low	~~S~~	~~R~~	~~M~~	1, 2, 3
e. Automatic Actuation Logic	N.A.	N.A.	~~M~~(1)	1, 2, 3
4. MAIN STEAM LINE ISOLATION				
a. Manual (Trip Buttons)	N.A.	N.A.	~~R~~	N.A.
b. Containment Pressure - High	~~S~~	~~R~~	~~M~~	1, 2, 3
c. Steam Generator Pressure - Low	~~S~~	~~R~~	~~M~~	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	~~M~~(1)	1, 2, 3
5. ENCLOSURE BUILDING FILTRATION (EBFAS)				
a. Manual EBFAS (Trip Buttons)	N.A.	N.A.	~~R~~	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	~~R~~	N.A.
c. Containment Pressure - High	~~S~~	~~R~~	~~M~~	1, 2, 3
d. Pressurizer Pressure - Low	~~S~~	~~R~~	~~M~~	1, 2, 3
e. Automatic Actuation Logic	N.A.	N.A.	~~M~~(1)	1, 2, 3

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. CONTAINMENT SUMP RECIRCULATION (SRAS)				
a. Manual SRAS (Trip Buttons)	N.A.	N.A.	~~R~~	N.A.
b. Refueling Water ~~Storage~~ Tank - Low	~~S~~	~~R~~	~~M~~	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	~~M~~(1)	1, 2, 3
7. DELETED				
8. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage - level one	~~S~~	~~R~~	~~M~~	1, 2, 3
b. 4.16 kv Emergency Bus Undervoltage - level two	~~S~~	~~R~~	~~M~~	1, 2, 3
9. AUXILIARY FEEDWATER				
a. Manual	N.A.	N.A.	~~R~~	N.A.
b. Steam Generator Level - Low	~~S~~	~~R~~	~~M~~	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	~~M~~	1, 2, 3
10. STEAM GENERATOR BLOWDOWN				
a. Steam Generator Level - Low	~~S~~	~~R~~	~~M~~	1, 2, 3

TABLE 4.3-2 (Continued)

the frequency specified in the Surveillance Frequency Control Program

TABLE NOTATION

(1) The coincident logic circuits shall be tested automatically or manually ~~at least once per 31 days~~. The automatic test feature shall be verified OPERABLE at ~~least once per 31 days~~. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or other specified conditions for surveillance testing of the following:

a. Pressurizer Pressure Safety Injection Automatic Actuation Logic; and

b. Pressurizer Pressure Containment Isolation Automatic Actuation Logic; and

c. Steam Generator Pressure Main Steam Line Isolation Automatic Actuation Logic; and

d. Pressurizer Pressure Enclosure Building Filtration Automatic Actuation Logic.

Testing of the automatic actuation logic for Pressurizer Pressure Safety Injection, Pressurizer Pressure Containment Isolation, and Pressurizer Pressure Enclosure Building Filtration shall be performed within 12 hours after exceeding a pressurizer pressure of 1850 psia in MODE 3. Testing of the automatic actuation logic for Steam Generator Pressure Main Steam Line Isolation shall be performed within 12 hours after exceeding a steam generator pressure of 700 psia in MODE 3.

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM SENSOR CABINET POWER SUPPLY DRAWERS

LIMITING CONDITION FOR OPERATION

3.3.2.2 The engineered safety feature actuation system Sensor Cabinets (RC02A1, RC02B2, RC02C3 & RC02D4) Power Supply Drawers shall be OPERABLE and energized from the normal power source with the backup power source available. The normal and backup power sources for each sensor cabinet is detailed in Table 3.3-5a:

CABINET	NORMAL POWER	BACKUP POWER
RC02A1	VA-10	VA-40
RC02B2	VA-20	VA-30
RC02C3	VA-30	VA-20
RC02D4	VA-40	VA-10

Table 3.3-5a

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With any of the Sensor Cabinet Power Supply Drawers inoperable, or either the normal or backup power source not available as delineated in Table 3.3-5a, restore the inoperable Sensor Cabinet Power Supply Drawer to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

at the frequency specified in the Surveillance Frequency Control Program

4.3.2.2.1 The engineered safety feature actuation system Sensor Cabinet Power Supply Drawers shall be determined OPERABLE ~~once per shift~~ by visual inspection of the power supply drawer indicating lamps.

4.3.2.2.2 Verify the OPERABILITY of the Sensor Cabinet Power Supply auctioneering circuit at ~~least once per 18 months~~.

the frequency specified in the Surveillance Frequency Control Program

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 2 hours or declare the channel inoperable.

b. With the number of OPERABLE channels less than the number of MINIMUM CHANNELS OPERABLE in Table 3.3-6, take the ACTION shown in Table 3.3-6. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

required

4.3.3.1.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-3.

4.3.3.1.2 DELETED

4.3.3.1.3 Verify the response time of the control room isolation channel ~~at least once per 18 months~~.

the frequency specified in the Surveillance Frequency Control Program

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

TABLE 4.3-3
RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Deleted				
b. Control Room Isolation	~~S~~	~~R~~	~~M~~	ALL MODES
c. Containment High Range	~~S~~	~~R~~*	~~M~~	1, 2, 3, & 4
2. PROCESS MONITORS				
a. Containment Atmosphere-Particulate	~~S~~	~~R~~	~~M~~	1, 2, 3, & 4
b. Deleted				
c. Noble Gas Effluent Monitor (high range) (Unit 2 Stack)	~~S~~	~~R~~	~~M~~	1, 2, 3, & 4

* Calibration of the sensor with a radioactive source need only be performed on the lowest range. Higher ranges may be calibrated electronically.

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the number of OPERABLE remote shutdown monitoring instrumentation channels less than required by Table 3.3-9, either:

a. Restore the inoperable channel to OPERABLE status within 7 days, or

b. Be in HOT SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

required

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

Replace each marked through surveillance frequency in the Check and Calibration columns with "SFCP"

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Wide Range Logarithmic Neutron Flux	~~M~~	~~R~~*
2.	Reactor Trip Breaker Indication	~~M~~	N.A.
3.	Reactor Cold Leg Temperature	~~M~~	~~R~~
4.	Pressurizer Pressure		
	a. Low Range	~~M~~	~~R~~
	b. High Range	~~M~~	~~R~~
5.	Pressurizer Level	~~M~~	~~R~~
6.	Steam Generator Level	~~M~~	~~R~~
7.	Steam Generator Pressure	~~M~~	~~R~~

* Neutron detectors are excluded from the CHANNEL CALIBRATION.

LIMITING CONDITION FOR OPERATION

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. ACTIONS per Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each required accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

Replace each marked through surveillance frequency in the Check and Calibration columns with "SFCP"

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Pressurizer Water Level	~~M~~	~~R~~
2.	Auxiliary Feedwater Flow Rate	~~M~~	~~R~~
3.	Reactor Coolant System Subcooled/Superheat Monitor	~~M~~	~~R~~
4.	PORV Position Indicator	~~M~~	~~R~~
5.	PORV Block Valve Position Indicator	N.A.	~~R~~
6.	Safety Valve Position Indicator	~~M~~	~~R~~
7.	Containment Pressure	~~M~~	~~R~~
8.	Containment Water Level (Narrow Range)	~~M~~	~~R~~
9.	Containment Water Level (Wide Range)	~~M~~	~~R~~
10.	Core Exit Thermocouples	~~M~~	~~R~~*
11.	Main Steam Line Radiation Monitor	~~M~~	~~R~~
12.	Reactor Vessel Coolant Level	~~M~~	~~R~~*

* Electronic calibration from the ICC cabinets only.

COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the requirements of the above specification not met, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

the frequency specified in the Surveillance Frequency Control Program

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 Two reactor coolant loops shall be OPERABLE and one reactor coolant loop shall be in operation.

NOTE

All reactor coolant pumps may not be in operation for up to 1 hour per 8 hour period provided:

a. no operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1.1; and

b. core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 3.

ACTION:

a. With one reactor coolant loop inoperable, restore the required reactor coolant loop to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop OPERABLE or in operation, immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1 and immediately initiate corrective action to return one required reactor coolant loop to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

at the frequency specified in the Surveillance Frequency Control Program

4.4.1.2.1 The required reactor coolant pump, if not in operation, shall be determined to be OPERABLE ~~once per 7 days~~ by verifying correct breaker alignment and indicated power available.

4.4.1.2.2 One reactor coolant loop shall be verified to be in operation at ~~least once per 12 hours~~.

4.4.1.2.3 Each steam generator secondary side water level shall be verified to be ≥ 10% narrow range at ~~least once per 12 hours~~.

the frequency specified in the Surveillance Frequency Control Program

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required pump, if not in operation, shall be determined OPERABLE ~~once per 7 days~~ by verifying correct breaker alignment and indicated power available.

at the frequency specified in the Surveillance Frequency Control Program

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE, by verifying the secondary side water level to be ≥ 10% narrow range at ~~least once per 12 hours~~.

4.4.1.3.3 One reactor coolant loop or shutdown cooling train shall be verified to be in operation at ~~least once per 12 hours~~.

the frequency specified in the Surveillance Frequency Control Program

LIMITING CONDITION FOR OPERATION (continued)

APPLICABILITY: MODE 5 with Reactor Coolant System loops filled.

ACTION: a. With one shutdown cooling train inoperable and any steam generator secondary water level not within limits, immediately initiate action to either restore a second shutdown cooling train to OPERABLE status or restore steam generator secondary water levels to within limit.

b. With no shutdown cooling train OPERABLE or in operation, immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1 and immediately initiate action to restore one shutdown cooling train to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

at the frequency specified in the Surveillance Frequency Control Program

4.4.1.4.1 The required shutdown cooling pump, if not in operation, shall be determined OPERABLE ~~once per 7 days~~ ~~by verifying~~ correct breaker alignment and indicated power available.

4.4.1.4.2 The required steam generators shall be determined OPERABLE, by verifying the secondary side water level to be ≥ 10% narrow range at ~~least once per 12 hours~~.

4.4.1.4.3 One shutdown cooling train shall be verified to be in operation at ~~least once per 12 hours~~.

the frequency specified in the Surveillance Frequency Control Program

REACTOR COOLANT SYSTEM

COOLANT LOOPS AND COOLANT CIRCULATION

COLD SHUTDOWN - REACTOR COOLANT SYSTEM LOOPS NOT FILLED

SURVEILLANCE REQUIREMENTS

at the frequency specified in the Surveillance Frequency Control Program

4.4.1.5.1 The required shutdown cooling pump, if not in operation, shall be determined OPERABLE ~~once per 7 days~~ by verifying correct breaker alignment and indicated power available.

4.4.1.5.2 One shutdown cooling train shall be verified to be in operation at ~~least once per 12 hours~~.

the frequency specified in the Surveillance Frequency Control Program

REACTOR COOLANT SYSTEM

REACTOR COOLANT PUMPS

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.6 A maximum of two reactor coolant pumps shall be OPERABLE.

APPLICABILITY: MODE 5

ACTION:

With more than two reactor coolant pumps OPERABLE, take immediate action to comply with Specification 3.4.1.6.

SURVEILLANCE REQUIREMENTS

4.4.1.6 Two reactor coolant pumps shall be demonstrated inoperable at ~~least once per 12 hours~~ the frequency specified in the Surveillance Frequency Control Program by verifying that the motor circuit breakers have been disconnected from their electrical power supply circuits.

SURVEILLANCE REQUIREMENTS

4.4.3.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE:

a. ~~Once per 31 days~~ by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. ~~Once per 18 months~~ by performance of a CHANNEL CALIBRATION.

c. ~~Once per 18 months~~ by operating the PORV through one complete cycle of full travel at conditions representative of MODES 3 or 4.

4.4.3.2 Each block valve shall be demonstrated OPERABLE ~~once per 92 days~~ by operating the valve through one complete cycle of full travel. This demonstration is not required if a PORV block valve is closed and power removed to meet Specification 3.4.3 b or c.

at the frequency specified in the Surveillance Frequency Control Program

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 70%, and

b. At least two groups of pressurizer heaters each having a capacity of at least 130 kW.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 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

4.4.4.1 The pressurizer water level shall be determined to be within its limits at ~~least once per 12 hours.~~

4.4.4.2 Verify at least two groups of pressurizer heaters each have a capacity of at least 130 kW at ~~least once per 92 days~~.

the frequency specified in the Surveillance Frequency Control Program

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

2. Appropriate grab samples of the containment atmosphere are obtained and analyzed for particulate radioactivity within 6 hours and at least once per 6 hours thereafter, and

3. A Reactor Coolant System water inventory balance is performed within 6 hours and at least once per 6 hours thereafter.

Otherwise, be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment sump level monitoring system-performance of CHANNEL CALIBRATION TEST at ~~least once per 18 months~~.

the frequency specified in the Surveillance Frequency Control Program

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. 75 GPD primary to secondary LEAKAGE through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any RCS operational LEAKAGE not within limits for reasons other than PRESSURE BOUNDARY LEAKAGE or primary to secondary LEAKAGE, reduce LEAKAGE to within limits within 4 hours.

b. With ACTION and associated completion time of ACTION a. not met, or PRESSURE BOUNDARY LEAKAGE exists, or primary to secondary LEAKAGE not within limits, be in HOT STANDBY within 6 hours and be in COLD SHUTDOWN within 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1

- - - - - - - - - - - - - - - - NOTES - - - - - - - - - - - - - - - -

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

2. Not applicable to primary to secondary LEAKAGE.

- -

Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance at ~~least once per 72 hours~~.

the frequency specified in the Surveillance Frequency Control Program

REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.2.2

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

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

Verify primary to secondary LEAKAGE is ≤ 75 gallons per day through any one SG at ~~least once per 72 hours~~.

the frequency specified in the Surveillance Frequency Control Program

4.4.8.1 Verify the specific activity of the primary coolant ≤ 1100 μCi/gram DOSE EQUIVALENT XE-133 ~~once per 7 days~~.*

4.4.8.2 Verify the specific activity of the primary coolant ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131 ~~once per 14 days~~,* and between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RATED THERMAL POWER within a one hour period.

at the frequency specified in the Surveillance Frequency Control Program

* Surveillance only required to be performed for MODE 1 operation, consistent with the provisions of Specification 4.0.1.

SURVEILLANCE REQUIREMENT

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at ~~least once per 31 days~~ thereafter when the PORV is required OPERABLE.

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

c. Verifying the PORV block valve is open at ~~least once per 72 hours~~ when the PORV is being used for overpressure protection.

d. Testing in accordance with the inservice test requirements of Specification 4.0.5.

4.4.9.3.2 Verify no more than the maximum allowed number of charging pumps are capable of injecting into the RCS at ~~least once per 12 hours~~.

4.4.9.3.3 Verify no more than the maximum allowed number of HPSI pumps are capable of injecting into the RCS at ~~least once per 12 hours~~.

4.4.9.3.4 Verify the required RCS vent is open at ~~least once per 31 days~~ when the vent pathway is provided by vent valve(s) that is(are) locked, sealed, or otherwise secured in the open position, otherwise, verify the vent pathway at ~~least once per 12 hours~~.

the frequency specified in the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

4.5.1 Each SIT shall be demonstrated OPERABLE:

a. Verify each SIT isolation valve is fully open at ~~least once per 12 hours~~.*(1)

b. Verify borated water volume in each SIT is ≥ 1080 cubic feet and ≤ 1190 cubic feet at ~~least once per 12 hours~~.**(2)

c. Verify nitrogen cover-pressure in each SIT is ≥ 200 psig and ≤ 250 psig at ~~least once per 12 hours~~.***(3)

d. Verify boron concentration in each SIT is ≥ 1720 ppm at ~~least once per 6 months,~~ and once within 6 hours after each solution volume increase of ≥ 1% of tank volume****(4) that is not the result of addition from the refueling water storage tank.

e. Verify that the closing coil in the valve breaker cubicle is removed at ~~least once per 31 days~~.

the frequency specified in the Surveillance Frequency Control Program

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*(1) If one SIT is inoperable, except as a result of boron concentration not within limits or inoperable level or pressure instrumentation, surveillance is not applicable to the affected SIT.

**(2) If one SIT is inoperable due solely to inoperable water level instrumentation, surveillance is not applicable to the affected SIT.

***(3) If one SIT is inoperable due solely to inoperable pressure instrumentation, surveillance is not applicable to affected SIT.

****(4)Only required to be performed for affected SIT.

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At ~~least once per 31 days~~ by verifying each Emergency Core Cooling System manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At ~~least once per 31 days~~ by verifying that the following valves are in the indicated position with power to the valve operator removed:

Valve Number	Valve Function	Valve Position
2-SI-306	Shutdown Cooling Flow Control	Open*
2-SI-659	SRAS Recirc.	Open**
2-SI-660	SRAS Recirc.	Open**

* Pinned and locked at preset throttle open position.

** To be closed prior to recirculation following LOCA.

c. By verifying the developed head of each high pressure safety injection pump at the flow test point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

d. By verifying the developed head of each low pressure safety injection pump at the flow test point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

e. By verifying the delivered flow of each charging pump at the required discharge pressure is greater than or equal to the required flow when tested pursuant to Specification 4.0.5.

f. At ~~least once per 18 months~~ by verifying each Emergency Core Cooling System automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

g. At ~~least once per 18 months~~ by verifying each high pressure safety injection pump and low pressure safety injection pump starts automatically on an actual or simulated actuation signal.

the frequency specified in the Surveillance Frequency Control Program

h. At ~~least once per 18 months~~ by verifying each low pressure safety injection pump stops automatically on an actual or simulated actuation signal.

i. By verifying the correct position of each electrical and/or mechanical position stop for each injection valve in Table 4.5-1:

1. Within 4 hours after completion of valve operations.

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

j. At ~~least once per 18 months~~ by verifying through visual inspection of the containment sump that each Emergency Core Cooling System subsystem suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.

k. At ~~least once per 18 months~~ by verifying the Shutdown Cooling System open permissive interlock prevents the Shutdown Cooling System inlet isolation valves from being opened with an actual or simulated Reactor Coolant System pressure signal of ≥ 300 psia.

the frequency specified in the Surveillance Frequency Control Program

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank shall be OPERABLE with:

a. A minimum contained volume of 370,000 gallons of borated water,

b. A minimum boron concentration of 1720 ppm,

c. A minimum water temperature of 50°F when in MODES 1 and 2, and

d. A minimum water temperature of 35°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore tank to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

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

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

b. When in MODES 3 and 4, at ~~least once per 24 hours~~ by verifying the RWST temperature is ≥ 35°F when the RWST ambient air temperature is < 35°F.

c. When in MODES 1 and 2, at ~~least once per 24 hours~~ by verifying the RWST temperature is ≥ 50°F when the RWST ambient air temperature is < 50°F.

the frequency specified in the Surveillance Frequency Control Program

TRISODIUM PHOSPHATE (TSP)

LIMITING CONDITION FOR OPERATION

3.5.5 The TSP baskets shall contain ≥282 ft^3 of active TSP.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With the quantity of TSP less than required, restore the TSP quantity within 72 hours, or be in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.5.1 Verify that the TSP baskets contain ≥282 ft^3 of TSP at ~~least once per 18 months~~.

4.5.5.2 Verify that a sample from the TSP baskets provides adequate pH adjustment of borated water at ~~least once per 18 months~~.

the frequency specified in the Surveillance Frequency Control Program

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 next 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At ~~least once per 31 days~~ by verifying that all penetrations[1] not capable of being closed by OPERABLE containment automatic isolation valves[2] and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions,[3] except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

b. At ~~least once per 31 days~~ by verifying the equipment hatch is closed and sealed.

c. By verifying the containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. After each closing of a penetration subject to type B testing (except the containment air lock), if opened following a Type A or B test, by leak rate testing in accordance with the Containment Leakage Rate Testing Program.

e. By verifying Containment structural integrity in accordance with the Containment Tendon Surveillance Program.

the frequency specified in the Surveillance Frequency Control Program

(1) 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 prior to entering MODE 4 from MODE 5, if not performed within the previous 92 days.

(2) In MODE 4, the requirement for an OPERABLE containment automatic isolation valve system is satisfied by use of the containment isolation trip pushbuttons

(3) Isolation devices in high radiation areas may be verified by use of administrative means.

CONTAINMENT AIR LOCKS

SURVEILLANCE REQUIREMENTS

4.6.1.3.1 Each containment air lock shall be demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program. Containment air lock leakage test results shall be evaluated against the leakage limits of Technical Specification 3.6.1.2. (An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test) .

4.6.1.3.2 Each containment air lock shall be demonstrated OPERABLE ~~at least once per 24 months~~ at the frequency specified in the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -12 inches Water Gauge and +1.0 PSIG.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure in excess of or below the limits above, restore the internal pressure to within the limits within 1 hour or be in HOT STANDBY within the next 4 hours; go to COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

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

the frequency specified in 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 > 120°F, reduce the average air temperature to within the limit within 8 hours, or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be determined to be ≤ 120°F at ~~least once per 24 hours.~~

the frequency specified in the Surveillance Frequency Control Program

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains, with each cooling train consisting of two containment air recirculation and cooling units, shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

Inoperable Equipment	Required ACTION
a. One containment spray train	a.1 Restore the inoperable containment spray train to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1750 psia within the following 6 hours.
b. One containment cooling train	b.1 Restore the inoperable containment cooling train to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.
c. One containment spray train AND One containment cooling train	c.1 Restore the inoperable containment spray train or the inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
d. Two containment cooling trains	d.1 Restore at least one inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
e. All other combinations	e.1 Enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

4.6.2.1.1 Each containment spray train shall be demonstrated OPERABLE:

a. At ~~least once per 31 days by~~ the frequency specified in the Surveillance Frequency Control Program verifying each containment spray manual, power operated, and automatic valve in the spray train flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.

* The Containment Spray System is not required to be OPERABLE in MODE 3 if pressurizer pressure is < 1750 psia.

the frequency specified in the Surveillance Frequency Control Program

b. By verifying the developed head of each containment spray pump at the flow test point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

c. At ~~least once per 18 months~~ by verifying each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

d. At ~~least once per 18 months~~ by verifying each containment spray pump starts automatically on an actual or simulated actuation signal.

e. By verifying each spray nozzle is unobstructed following activities that could cause nozzle blockage.

4.6.2.1.2 Each containment air recirculation and cooling unit shall be demonstrated OPERABLE:

a. At ~~least once per 31 days~~ by operating each containment air recirculation and cooling unit in slow speed for $\geq$ 15 minutes.

b. At ~~least once per 31 days~~ by verifying each containment air recirculation and cooling unit cooling water flow rate is $\geq$ 500 gpm.

c. At ~~least once per 18 months~~ by verifying each containment air recirculation and cooling unit starts automatically on an actual or simulated actuation signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE.(1) (2)

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, either:

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

b. Isolate the affected penetration(s) within 4 hours by use of a deactivated automatic valve(s) secured in the isolation position(s), or

c. Isolate the affected penetration(s) within 4 hours by use of a closed manual valve(s) or blind flange(s); or

d. Isolate the affected penetration that has only one containment isolation valve and a closed system within 72 hours by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange; or

e. Be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each containment isolation valve shall be demonstrated OPERABLE:

a. By verifying the isolation time of each power operated automatic containment isolation valve when tested pursuant to Specification 4.0.5.

b. At ~~least once per 18 months~~ the frequency specified in the Surveillance Frequency Control Program by verifying each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.

(1) Containment isolation valves may be opened on an intermittent basis under administrative controls.

(2) The provisions of this Specification in MODES 1, 2 and 3, are not applicable for main steam line isolation valves. However, provisions of Specification 3.7.1.5 are applicable for main steam line isolation valves.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3.2 The containment purge supply and exhaust isolation valves shall be sealed closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment purge supply and/or one exhaust isolation valve open, close the open valve(s) 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

4.6.3.2 The containment purge supply and exhaust isolation valves shall be determined sealed closed at ~~least once per 31 days~~.

the frequency specified in the Surveillance Frequency Control Program

POST-INCIDENT RECIRCULATION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.4.4 Two separate and independent post-incident recirculation systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one post-incident recirculation system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

the frequency specified in the Surveillance Frequency Control Program

4.6.4.4 Each post-incident recirculation system shall be demonstrated OPERABLE at ~~least once per 92 days on a STAGGERED TEST BASIS~~ by:

a. Verifying that the system can be started on operator action in the control room, and

b. Verifying that the system operates for at least 15 minutes.

3/4.6.5 SECONDARY CONTAINMENT

ENCLOSURE BUILDING FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two separate and independent Enclosure Building Filtration Trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one Enclosure Building Filtration Train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

the frequency specified in the Surveillance Frequency Control Program

4.6.5.1 Each Enclosure Building Filtration Train shall be demonstrated OPERABLE:

a. At ~~least once per 31 days on a STAGGERED TEST BASIS~~ by initiating, from the control room, flow through the HEPA filter and charcoal absorber train and verifying that the train operates for at least 10 hours with the heaters on.

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

1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 9000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

3. Verifying a train flow rate of 9000 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

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

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is ≤ 2.6 inches Water Gauge while operating the train at a flow rate of 9000 cfm ± 10%.

2. Verifying that the train starts on an Enclosure Building Filtration Actuation Signal (EBFAS).

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 9000 cfm ± 10%.

the frequency specified in the Surveillance Frequency Control Program

* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89. Additionally, the charcoal sample shall have a removal efficiency of ≥ 95%.

ENCLOSURE BUILDING

LIMITING CONDITION FOR OPERATION

3.6.5.2 The Enclosure Building shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the Enclosure Building inoperable, restore the Enclosure Building to OPERABLE status within 24 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2.1 OPERABILITY of the Enclosure Building shall be demonstrated at ~~least once per 31 days~~ by verifying that each access opening is closed except when the access opening is being used for normal transit entry and exit.

4.6.5.2.2. At ~~least once per 18 months~~ verify each Enclosure Building Filtration Train produces a negative pressure of greater than or equal to 0.25 inches W.G. in the Enclosure Building Filtration Region within 1 minute after an Enclosure Building Filtration Actuation Signal.

the frequency specified in the Surveillance Frequency Control Program

AUXILIARY FEEDWATER PUMPS

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

Inoperable Equipment	Required ACTION
e. Three auxiliary feedwater pumps in MODE 1, 2, or 3.	e. - - - - - - - - NOTE - - - - - - - - LCO 3.0.3 and all other LCO required ACTIONS requiring MODE changes are suspended until one AFW pump is restored to OPERABLE status. - Immediately initiate ACTION to restore one auxiliary feedwater pump to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At ~~least once per 31 days~~ by verifying each auxiliary feedwater manual, power operated, and automatic valve in each water flow path and in each steam supply flow path to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. By verifying the developed head of each auxiliary feedwater pump at the flow test point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5. (Not required to be performed for the steam turbine driven auxiliary feedwater pump until 24 hours after reaching 800 psig in the steam generators. The provisions of Specification 4.0.4 are not applicable to the steam turbine driven auxiliary feedwater pump for entry into MODE 3.)

the frequency specified in the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (Continued)

c. At ~~least once per 18 months~~ by verifying each auxiliary feedwater automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position, as designed, on an actual or simulated actuation signal.

d. At ~~least once per 18 months~~ by verifying each auxiliary feedwater pump starts automatically, as designed, on an actual or simulated actuation signal.

e. By verifying the proper alignment of the required auxiliary feedwater flow paths by verifying flow from the condensate storage tank to each steam generator prior to entering MODE 2 whenever the unit has been in MODE 5, MODE 6, or defueled for a cumulative period of greater than 30 days.

the frequency specified in the Surveillance Frequency Control Program

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank shall be OPERABLE with a minimum contained volume of 165,000 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With less than 165,000 gallons of water in the condensate storage tank, within 4 hours either:

a. Restore the water volume to within the limit or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the fire water system as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank water volume to within its limits within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tank shall be demonstrated OPERABLE at ~~least once per 12 hours~~ by verifying the water level.

the frequency specified in the Surveillance Frequency Control Program

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM FREQUENCY
1. Gross Activity Determination	~~3 times per 7 days~~ with a maximum time of 72 hours between samples.
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.

At the frequency specified in the Surveillance Frequency Control Program

b. With two or more of the feedwater isolation components inoperable in the same flow path, either:

1. Restore the inoperable component(s) to OPERABLE status within 8 hours until ACTION 'a' applies, or

2. Isolate the affected flow path within 8 hours, and verify that the inoperable feedwater isolation components are closed or isolated/secured once per 7 days, or

3. Be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

the frequency specified in the Surveillance Frequency Control Program

4.7.1.6 Each feedwater isolation valve/feedwater pump trip circuitry shall be demonstrated OPERABLE at ~~least once per 18 months~~ by:

a. Verifying that on 'A' main steam isolation test signal, each isolation valve actuates to its isolation position, and

b. Verifying that on 'B' main steam isolation test signal, each isolation valve actuates to its isolation position, and

c. Verifying that on 'A' main steam isolation test signal, each feedwater pump trip circuit actuates, and

d. Verifying that on 'B' main steam isolation test signal, each feedwater pump trip circuit actuates.

ATMOSPHERIC DUMP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.7 Each atmospheric dump valve line shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one atmospheric dump valve line inoperable, restore the inoperable line to OPERABLE status within 48 hours or be in MODE 3 within the next 6 hours and MODE 4 within the following 24 hours.

b. With more than one atmospheric dump valve line inoperable, restore one inoperable line to OPERABLE status within 1 hour or be in MODE 3 within the next 6 hours and MODE 4 within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.7 Verify the OPERABILITY of each atmospheric dump valve line by local manual operation of each valve in the flowpath through one complete cycle of operation at ~~least once per 18 months~~.

the frequency specified in the Surveillance Frequency Control Program

STEAM GENERATOR BLOWDOWN ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.8 Each steam generator blowdown isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With one or more steam generator blowdown isolation valves inoperable, either:

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

b. Isolate the affected steam generator blowdown line within 4 hours; or

c. Be in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.8 Verify the closure time of each steam generator blowdown isolation valve is ≤ 10 seconds on an actual or simulated closure signal at ~~least once per 18 months~~.

the frequency specified in the Surveillance Frequency Control Program

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 Two reactor building closed cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one reactor building closed cooling water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 Each reactor building closed cooling water loop shall be demonstrated OPERABLE:

a. At ~~least once per 31 days~~ by verifying each reactor building closed cooling water manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At ~~least once per 18 months~~ by verifying each reactor building closed cooling water automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

c. At ~~least once per 18 months~~ by verifying each reactor building closed cooling water pump starts automatically on an actual or simulated actuation signal.

at the frequency specified in the Surveillance Frequency Control Program

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 Two service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one service water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

a. At ~~least once per 31 days~~ by verifying each service water manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At ~~least once per 18 months~~ by verifying each service water automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

c. At ~~least once per 18 months~~ by verifying each service water pump starts automatically on an actual or simulated actuation signal.

at the frequency specified in the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

the frequency specified in the Surveillance Frequency Control Program

4.7.6.1 Each Control Room Emergency Ventilation Train shall be demonstrated OPERABLE:

a. At ~~least once per 12 hours by verifying~~ that the control room air temperature is ≤ 100°F.

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

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

1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 2500 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accor-dance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revi-sion 2, March 1978.* The carbon sample shall have a removal efficiency of ≥ 95 percent.

3. Verifying a train flow rate of 2500 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89.

SURVEILLANCE REQUIREMENTS (Continued)

the frequency specified in the Surveillance Frequency Control Program

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

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3.4 inches Water Gauge while operating the train at a flow rate of 2500 cfm ± 10%.

2. Verifying that on a recirculation signal, with the Control Room Emergency Ventilation Train operating in the normal mode and the smoke purge mode, the train automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3/4.7.11 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.11 The ultimate heat sink shall be OPERABLE with a water temperature of less than or equal to 75°F.

APPLICABILITY: MODES 1, 2, 3, AND 4

ACTION:

a. With the ultimate heat sink water temperature > 75°F and ≤ 77°F, operation may continue provided the water temperature averaged over the previous 24 hour period is verified ≤75°F at least once per hour. Otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the ultimate heat sink water temperature > 77°F, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.11 The ultimate heat sink shall be determined OPERABLE:

a. At ~~least once per 24 hours~~ by verifying the water temperature to be within limits.

b. At ~~least once per 6 hours~~ by verifying the water temperature to be within limits when the water temperature exceeds 70°F.

the frequency specified in the Surveillance Frequency Control Program

ACTION (Continued)

Inoperable Equipment	Required ACTION
e. Two diesel generators	e.1 Perform Surveillance Requirement 4.8.1.1.1 for the offsite circuits within 1 hour and at least once per 8 hours thereafter. AND e.2 Restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. AND e.3 Following restoration of one diesel generator restore remaining inoperable diesel generator to OPERABLE status following the time requirements of ACTION Statement b above based on the initial loss of the remaining inoperable diesel generator.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Verify correct breaker alignment and indicated power available for each required offsite circuit at ~~least once per 24 hours~~.

the frequency specified in the Surveillance Frequency Control Program

the frequency specified in the Surveillance Frequency Control Program

4.8.1.1.2 Each required diesel generator shall be demonstrated OPERABLE:*

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

1. Verifying the fuel level in the fuel oil supply tank,

2.

NOTES
1. A modified diesel generator start involving idling and gradual acceleration to synchronous speed may be used as recommended by the manufacturer. When modified start procedures are not used, the requirements of SR 4.8.1.1.2.d.1 must be met.
2. Performance of SR 4.8.1.1.2.d satisfies this Surveillance Requirement.

Verifying the diesel generator starts from standby conditions and achieves steady state voltage ≥ 3740 V and ≤ 4580 V, and Frequency ≥ 58.8 Hz and ≤ 61.2 Hz.

3.

NOTES
1. Diesel generator loading may include gradual loading as recommended by the manufacturer.
2. Momentary transients outside the load range do not invalidate this test.
3. This test shall be conducted on only one diesel generator at a time.
4. This test shall be preceded by and immediately follow without shutdown a successful performance of SR 4.8.1.1.2.a.2, or SRs 4.8.1.1.2.d.1 and 4.8.1.1.2.d.2.
5. Performance of SR 4.8.1.1.2.d satisfies this Surveillance Requirement.

Verifying the diesel generator is synchronized and loaded, and operates for ≥ 60 minutes at a load ≥ 2475 kW and ≤ 2750 kW.

* All diesel starts may be preceded by an engine prelube period.

SURVEILLANCE REQUIREMENTS (Continued)

b. The diesel fuel oil supply shall be checked by:

1. Checking for and removing accumulated water from each fuel oil storage tank at ~~least once per 92 days~~.

2. Verifying fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program in accordance with the Diesel Fuel Oil Testing Program.

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

the frequency specified in the Surveillance Frequency Control Program

1. Deleted

2.

NOTE

This surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced.

Verifying that the automatic time delay sequencer is OPERABLE with the following settings:

Sequence Step	Time After Closing of Diesel Generator Output Breaker (Seconds)	
	Minimum	Maximum
1 (T_1)	1.5	2.2
2 (T_2)	$T_1 + 5.5$	8.4
3 (T_3)	$T_2 + 5.5$	14.6
4 (T_4)	$T_3 + 5.5$	20.8

SURVEILLANCE REQUIREMENT (Continued)

the frequency specified in the Surveillance Frequency Control Program

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

1. Verifying the diesel starts from standby conditions and accelerates to $\geq$ 90% of rated speed and to $\geq$ 97% of rated voltage within 15 seconds after the start signal.

2. Verifying the generator achieves steady state voltage $\geq$ 3740 V and $\leq$ 4580 V, and frequency $\geq$ 58.8 Hz and $\leq$ 61.2 Hz.

3.

NOTES
1. Diesel generator loading may include gradual loading as recommended by the manufacturer.
2. Momentary transients outside the load range do not invalidate this test.
3. This test shall be conducted on only one diesel generator at a time.
4. This test shall be preceded by and immediately follow without shutdown a successful performance of SRs 4.8.1.1.2.d.1 and 4.8.1.1.2.d.2, or SR 4.8.1.1.2.a.2.

Verifying the diesel generator is synchronized and loaded, and operates for $\geq$ 60 minutes at a load $\geq$ 2475 kW and $\leq$ 2750 kW.

3/4 8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generators with tie breakers open between redundant busses:

4160	volt Emergency Bus # 24 C
4160	volt Emergency Bus #24 D
480	volt Emergency Load Center #22 E
480	volt Emergency Load Center #22 F
120	volt A.C. Vital Bus # VA-10
120	volt A.C. Vital Bus # VA-20
120	volt A.C. Vital Bus # VA-30
120	volt A.C. Vital Bus # VA-40

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus and/ or associated load center to OPERABLE status within 8 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from normal A.C. sources with tie breakers open between redundant busses at ~~least once per 7 days~~ by verifying correct breaker alignment and indicated power availability~~.~~

the frequency specified in the Surveillance Frequency Control Program

LIMITING CONDITION FOR OPERATION (Continued)

3.8.2.1A Inverters 5 and 6 shall be OPERABLE and available for automatic transfer via static switches VS1 and VS2 to power busses VA-10 and VA-20, respectively.

APPLICABILITY: MODES 1, 2 & 3

ACTION:

a. With inverter 5 or 6 inoperable, restore the inverter to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

b. With inverter 5 or 6 unavailable for automatic transfer via static switch VS1 or VS2 to power bus VA-10 or VA-20, respectively, restore the automatic transfer capability within 7 days or be in HOT SHUTDOWN within the next 12 hours.

c. With inverters 5 and 6 inoperable or unavailable for automatic transfer via static switches VS1 and VS2 to power busses VA-10 and VA-20, respectively, restore the inverters to OPERABLE status or restore their automatic transfer capability within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1A

a. Verify correct inverter voltage, frequency, and alignment for automatic transfer via static switches VS1 and VS2 to power busses VA-10 and VA-20, respectively, at ~~least once per 7 days~~.

b. Verify that busses VA-10 and VA-20 automatically transfer to their alternate power sources, inverters 5 and 6, respectively, at ~~least once per refueling~~ during shutdown.

the frequency specified in the Surveillance Frequency Control Program

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses shall be OPERABLE and energized from sources of power other than a diesel generator but aligned to an OPERABLE diesel generator:

1 - 4160 volt Emergency Bus

1 - 480 volt Emergency Load Center

2 - 120 volt A.C. Vital Busses

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, suspend all operations involving CORE ALTERATIONS and positive reactivity additions that could result in loss of required SDM or boron concentration, and movement of recently irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE and energized from normal A.C. sources at ~~least once per 7 days~~ by verifying correct breaker alignment and indicated power availability.

the frequency specified in the Surveillance Frequency Control Program

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 125-volt D.C. bus Train A and 125-volt D.C. bus Train B electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one 125-volt D.C. bus train inoperable, restore the inoperable 125-volt D.C. bus train to OPERABLE status within 2 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each 125-volt D.C. bus train shall be determined OPERABLE at ~~least once per 7 days~~ by verifying correct breaker alignment and indicated power availability.

4.8.2.3.2 Each 125-volt D.C. battery bank and charger of Train A and Train B shall be demonstrated OPERABLE:

a. By verifying ~~at least once per 7 days~~ that that the battery cell parameters meet Table 4.8-1 Category A limits.

b. By verifying ~~at least once per 92 days~~ the battery cell parameters meet Table 4.8-1 Category B limits.

the frequency specified in the Surveillance Frequency Control Program

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 deterioration that could degrade battery performance,

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

3. The battery charger will supply at least 400 amperes at a minimum of 130 volts for at least 12 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 emergency loads for 8 hours 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.

the frequency specified in the Surveillance Frequency Control Program

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 One 125 - volt D.C. bus train electrical power subsystem shall be OPERABLE:

APPLICABILITY: MODES 5 and 6.

ACTION:

With no 125-volt D.C. bus trains OPERABLE, suspend all operations involving CORE ALTERATIONS and positive reactivity additions that could result in loss of required SDM or boron concentration, and movement of recently irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus train shall be determined OPERABLE at ~~least once per 7 days~~ by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt D.C. bus train battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

the frequency specified in the Surveillance Frequency Control Program

D.C. DISTRIBUTION SYSTEMS (TURBINE BATTERY) — OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.5 The Turbine Battery 125-volt D.C. electrical power subsystem shall be OPERABLE.

APPLICABILITY: MODES 1, 2 & 3

ACTION:

a. With the Turbine Battery 125-volt D.C. electrical power subsystem inoperable, restore the subsystem to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

at the frequency specified in the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

4.8.2.5.1 Verify 125-volt D.C. bus 201D is OPERABLE at ~~least once per 7 days.~~

4.8.2.5.2 125-volt D.C. battery bank 201D shall be demonstrated OPERABLE:

a. By verifying at ~~least once per 7 days~~ that the battery cell parameters meet Table 4.8-2 Category A limits.

b. By verifying ~~at least once per 92 days~~ the battery cell parameters meet Table 4.8-2 Category B limits.

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 deterioration that could degrade battery performance, and

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

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 loads for 1 hour 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.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATIONS

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 sufficient to ensure that the more restrictive of 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 1720 ppm.

APPLICABILITY: MODE 6.

NOTE Only applicable to the refueling canal when connected to the Reactor Coolant System

ACTION:

With the requirements of the above specification not satisfied, within 15 minutes suspend all operations involving CORE ALTERATIONS and positive reactivity additions and initiate and continue boration at greater than or equal to 40 gpm of boric acid solution at or greater than the required refueling water storage tank concentration (ppm) 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 1720 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 CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of all filled portions of the reactor coolant system and the refueling canal shall be determined by chemical analysis at ~~least once per 72 hours~~ the frequency specified in the Surveillance Frequency Control Program.

4.9.1.3 Deleted

LIMITING CONDITION FOR OPERATION

3.9.2 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, immediately suspend all operations involving CORE ALTERATIONS and operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.

b. With both of the above required monitors inoperable, immediately initiate action to restore one monitor to OPERABLE status. Additionally, determine that the boron concentration of the Reactor Coolant System satisfies the requirements of LCO 3.9.1 within 4 hours and at least once per 12 hours thereafter.

SURVEILLANCE REQUIREMENTS.

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

a. Deleted

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

c. A CHANNEL CHECK and verification of audible counts ~~at least once per 12 hours~~.

at the frequency specified in the Surveillance Frequency Control Program

* Neutron detectors are excluded from CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Verify each required containment penetration is in the required status at ~~least once per 7 days~~ the frequency specified in the Surveillance Frequency Control Program.

4.9.4.2 Deleted

SHUTDOWN COOLING AND COOLANT CIRCULATION - HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

ACTION:

With no shutdown cooling train OPERABLE or in operation, perform the following actions:

a. Immediately suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1 and the loading of irradiated fuel assemblies in the core; and

b. Immediately initate action to restore one shutdown cooling train to OPERABLE status and operation; and

c. Within 4 hours place the containment penetrations in the following status:

   1. Close the equipment door and secure with at least four bolts; and

   2. Close at least one personnel airlock door; and

   3. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed with a manual or automatic isolation valve, blind flange, or equivalent.

SURVEILLANCE REQUIREMENTS

4.9.8.1 One shutdown cooling train shall be verified to be in operation and circulating reactor coolant at a flow rate greater than or equal to 1000 gpm at ~~least once per 12 hours~~.

the frequency specified in the Surveillance Frequency Control Program

LIMITING CONDITION FOR OPERATION (continued)

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed with a manual or automatic isolation valve, blind flange, or equivalent.

SURVEILLANCE REQUIREMENTS

the frequency specified in the Surveillance Frequency Control Program

4.9.8.2.1 One shutdown cooling train shall be verified to be in operation and circulating reactor coolant at a flow rate greater than or equal to 1000 gpm at ~~least once per 12 hours~~.

4.9.8.2.2 The required shutdown cooling pump, if not in operation, shall be determined OPERABLE ~~once per 7 days~~ by verifying correct breaker alignment and indicated power available.

at the frequency specified in the Surveillance Frequency Control Program

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.11 As a minimum, 23.0 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts.

During movement of irradiated fuel assemblies within containment.

ACTION:

With the water level less than that specified above, immediately suspend CORE ALTERATIONS and immediately suspend movement of irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level shall be determined to be within its minimum depth at ~~least once per 24 hours.~~

the frequency specified in the Surveillance Frequency Control Program

REFUELING OPERATIONS

STORAGE POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.12 As a minimum, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: WHENEVER IRRADIATED FUEL ASSEMBLIES ARE IN THE STORAGE POOL.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel and spent fuel pool platform crane operations with loads in the fuel storage areas.

SURVEILLANCE REQUIREMENTS

4.9.12 The water level in the storage pool shall be determined to be within its minimum depth at ~~least once per 7 days~~ when irradiated fuel assemblies are in the fuel storage pool.

the frequency specified in the Surveillance Frequency Control Program

SHIELDED CASK

LIMITING CONDITION FOR OPERATION

3.9.16.1 All fuel within a distance L from the center of the spent fuel pool cask laydown area shall have decayed for at least 1 year. The distance L equals the major dimension of the shielded cask.

APPLICABILITY: Whenever a shielded cask is on the refueling floor.

ACTION:

With the requirements of the above specification not satisfied, do not move a shielded cask to the refueling floor. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.16.1 The decay time of all fuel within a distance L from the center of the spent fuel pool cask laydown area shall be determined to be ≥ 1 year within 24 hours prior to moving a shielded cask to the refueling floor and at ~~least once per 72 hours~~ thereafter.

the frequency specified in the Surveillance Frequency Control Program

SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.17 The boron concentration in the spent fuel pool shall be greater than or equal to 1720 parts per million (ppm).

APPLICABILITY: Whenever any fuel assembly or consolidated fuel storage box, is stored in the spent fuel pool.

ACTION:

With the boron concentration less than 1720 ppm, suspend the movement of all fuel, consolidated fuel storage boxes, and shielded casks, and immediately initiate action to restore the spent fuel pool boron concentration to within its limit.

The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.17 Verify that the boron concentration is greater than or equal to 1720 ppm ~~every 7 days~~, and within 24 hours prior to the initial movement of a fuel assembly or consolidated fuel storage box in the Spent Fuel Pool, or shielded cask over the cask laydown area.

at the frequency specified in the Surveillance Frequency Control Program

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The requirement of Specifications 3.1.1.1, 3.1.3.5 and 3.1.3.6 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3[(1)] during PHYSICS TESTS.

ACTION:

a. With any CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, within 15 minutes initiate and continue boration at > 40 gpm of boric acid solution at or greater than the required refueling water storage tank (RWST) concentration (ppm) until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at > 40 gpm of boric acid solution at or greater than the required refueling water storage tank (RWST) concentration (ppm) until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTSS

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

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position once within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1 [(2)].

the frequency specified in the Surveillance Frequency Control Program

[(1)] Operation in MODE 3 shall be limited to 6 consecutive hours.

[(2)] Not required to be performed during initial power escalation following a refueling outage if SR 4.1.3.4 has been met

GROUP HEIGHT AND INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

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

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 are suspended, immediately:

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

b. Be in HOT STANDBY within 2 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at ~~least once per hour~~ during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 or 3.2.4 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 or 3.2.4 are suspended.

the frequency specified in the Surveillance Frequency Control Program

6.27 CONTROL ROOM ENVELOPE HABITABILITY PROGRAM (Continued)

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

f. The provisions of Surveillance Requirement 4.0.2 are applicable to the frequencies for assessing CRE habitability and determining CRE unfiltered inleakage as required by paragraph c.

Insert 1

6.28 SURVEILLANCE FREQUENCY CONTROL PROGRAM

This program provides controls for surveillance frequencies. The program shall ensure that surveillance requirements specified in the technical specification are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of frequencies of those surveillance requirements for which the frequency is controlled by the program.

b. Changes to the frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the frequencies established in the Surveillance Frequency Control Program.