ATTACHMENT 1

DESCRIPTION AND ASSESSMENT OF PROPOSED CHANGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

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 (TS) for Millstone Power Station Unit 3 (MPS3). 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 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 MPS3 and justify this amendment to incorporate the changes to the MPS3 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 MPS3 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 definition of STAGGERED TEST BASIS is being retained in MPS3 TS Definition Section 1 since this terminology is mentioned in Administrative TS Section 6.8.4.h, "Control Room Envelope Habitability Program," which is not the subject of this amendment request and is not proposed to be changed. This represents an administrative deviation from TSTF-425 with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

3. The inserts provided in TSTF-425 are revised to fit the MPS3 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 surveillance frequency is 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 MPS3 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 to the TSTF-425 inserts and the existing TS wording are made to fit the MPS3 TS format.

4. Attachment 4 provides a cross-reference between the NUREG-1431 surveillances included in TSTF-425 versus the MPS3 surveillances included in this amendment request. Attachment 4 includes a summary description of the referenced TSTF-425 (NUREG-1431)/MPS3 (NUREG-0452 format) 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-1431 surveillances included in TSTF-425 and corresponding MPS3 surveillances with plant-specific surveillance numbers,

b. NUREG-1431 surveillances included in TSTF-425 that are not contained in the MPS3 TS, and

c. MPS3 plant-specific surveillances that are not contained in NUREG-1431 and, therefore, are not included in the TSTF-425 mark-ups.

Concerning the above, MPS3 TSs were developed based on NUREG-0452. As a result, the applicable MPS3 TSs and associated Bases number differ from the STS presented in NUREG-1431 and TSTF-425, but with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

For NUREG-1431 surveillances not contained in MPS3 TSs, the corresponding mark-ups identified in TSTF-425 for these surveillances are not applicable. 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 MPS3 plant-specific surveillances not included in the NUREG-1431 mark-ups provided in TSTF-425, DNC has determined that since these surveillances involve fixed periodic frequencies, relocation of these frequencies is consistent with TSTF-425, Rev. 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.

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," (ADAMS Accession No. ML071360456), 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 MPS3, 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 MPS3.

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 MPS3, 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 3

DOCUMENTATION OF PROBABILISTIC RISK ASSESSMENT (PRA) TECHNICAL ADEQUACY

PRA QUALITY OVERVIEW

The implementation of the Surveillance Frequency Control Program (SFCP, also referred to as Technical Specifications Initiative 5b) at Millstone Power Station Unit 3 (MPS3) will follow the guidance provided in NEI 04-10, Revision 1 [Ref. 1] 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 STIs changes within the proposed licensee controlled program.

- Each proposed STI revision is reviewed to determine whether there are any commitments made to the Nuclear Regulatory Commission (NRC) that may prohibit changing the interval. If there are no related commitments, or the commitments may be changed using a commitment change process based on NRC endorsed guidance, then evaluation of the STI revision can proceed. If a commitment exists and the commitment change process does not permit the change without NRC approval, then the STI revision cannot be implemented. Only after receiving formal NRC approval to change the commitment could a STI revision proceed.

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

- Each STI revision is reviewed by an expert panel, referred to as the Integrated Decisionmaking Panel (IDP), which is normally the same panel as is used for Maintenance Rule implementation, but with the addition of specialists with experience in surveillance tests and system or component reliability. If the IDP approves the STI revision, the change is documented and implemented, and available for future audits by the NRC. If the IDP does not approve the STI revision, the STI value is left unchanged.

- Performance monitoring is conducted as recommended by the IDP. In some cases, no additional monitoring may be necessary beyond that already conducted under the Maintenance Rule. Performance monitoring helps to confirm that no failure mechanisms related to the revised test interval are subsequently identified as sufficiently significant to alter the basis provided in the justification for the surveillance interval change.

- 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 back to the previously acceptable STI.

- In addition to the above steps, the Probabilistic Risk Assessment (PRA) is used, when possible, to quantify the effect of a proposed individual STI revision compared to acceptance criteria in NEI 04-10, Revision 1. Also, the cumulative impact of risk-informed STI revisions on 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. The NEI 04-10, Revision 1 methodology endorses the guidance provided in Regulatory Guide (RG) 1.200, Revision 1 [Ref. 2], "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 (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 below 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 below.):

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, applicable modes), 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 (currently, RG 1.200, Revision 1,

includes only internal events PRA standard). 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.

The purpose of the remaining portion of this attachment is to address the requirements identified in Item 4 above.

TECHNICAL ADEQUACY OF THE PRA MODEL

The MPS3 PRA model of record, M308A, 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 M308A 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 MPS3 PRA is based on the event tree/fault tree methodology, which is a well-known methodology in the industry.

Dominion employs a structured approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Dominion nuclear generation 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 MPS3 PRA.

<u>PRA Maintenance and Update</u>

The MPS3 PRA model and documentation has been maintained as a living program. The PRA is routinely updated approximately every 3 to 5 years in order to reflect the current plant configuration and to reflect the accumulation of additional plant operating history and component failure data.

There are several procedures and GARDs (Guidance and Reference 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 any station in the Dominion Nuclear Fleet, including MPS3.
- 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 PRAs. Nevertheless, non-routine risk assessments are often unique, requiring departure from these guidelines in order to correctly perform and meet the risk assessment objectives.

A procedurally controlled process is used to maintain configuration control of the MPS3 PRA models, data, and software. In addition to model control, administrative mechanisms are in place to assure that plant modifications, procedure changes, calculations, operator training, system operation changes, and industry operating experiences (OEs) are appropriately screened, dispositioned, and scheduled for incorporation into the model. These processes help assure that the MPS3 PRA reflects the as-built, as-operated plant within the limitations of the PRA methodology.

This process involves a periodic review and update cycle to model any changes in the plant design or operation. Plant 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 modeling and/or documentation change is warranted. These reviews are documented, and if any PRA changes are warranted, they are added to the PRA Configuration Control (PRACC) database for PRA implementation tracking.

As part of the PRA evaluation for each STI change request, a review of open items in the PRACC database will be performed and an assessment of the impact on the results of the application will be made prior to presenting the results of the risk analysis to the 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 MPS3 PRACC database was reviewed to identify any open (i.e., not yet officially resolved and incorporated into the PRA) PRACC items. The open PRACC items contain identified PRA changes to address plant modifications (as discussed above) as well as changes to correct errors or to enhance the model.

The Level 1 and Level 2 MPS3 PRA analyses were originally developed and submitted to the NRC in 1983 as the Plant Safety Study (PSS). In response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities – 10 CFR 50.54(f)," the Millstone Unit No. 3 Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE) were submitted in the same letter to the NRC dated August 31, 1990 [Ref. 3]. The NRC staff evaluation reports for the IPE (May 5, 1992) [Ref. 4] and IPEEE (May 26, 1998) [Ref. 5] concluded that the studies meet the intent of Generic Letter 88-20. The MPS3 PRA has been updated many times since the original PSS. A summary of the MPS3 PRA history is listed below.

Date	Model Change
08/83	MPS3 PSS submitted
09/83	Amendment 1: Corrected consequence analysis
01/84	Transfer of PSS technology from Westinghouse, the PSS contractor, to the licensee
04/84	Amendment 2: Reanalysis of seismic fragilities by Structural Mechanics Associates
11/84	Amendment 3: Correction of mathematical error in seismic analysis
08/85	Published MPS3 risk evaluation report (NUREG-1152)
08/87	Amendment 4 (internal): Reanalysis of the Level 1 PRA to account for actual surveillance intervals, main feedwater recovery, etc.
1988	First round of evaluation of projects under internal Integrated Safety Assessment Program (ISAP)
1989	Second round of internal ISAP evaluations
89-90	Transferred PSS from mini-computer to personal computer
05/90	5th update: Correction of math and logic errors discovered in transfer
06/90	6th update: Updated transient frequencies (plant data), revised V sequence model, and coupled the Level 2 PRA to the Level 1
Fall 90	Coupled the Level 3 PRA to Levels 1 and 2; third round of ISAP evaluations
08/90	Submittal of IPE
05/92	NRC staff evaluation report concludes IPE meets the intent of Generic Letter 88-20. The report contains recommendations to explicitly model 1) total loss of service water (SW) initiating event, 2) Heating Ventilation and Air Conditioning (HVAC) dependency, and 3) Direct Current (DC) power dependency
12/95	Model converted from support state to linked fault tree methodology a. HVAC dependency explicitly modeled b. DC power dependency explicitly modeled c. Total loss of SW initiator modeled
02/96	LERF model developed using original PSS model
10/98	Station Blackout (SBO) diesel generator battery limitation modeled a. Transfer to sump recirculation analyzed using simulator data b. Plant-specific data update
08/99	Time-dependent SBO model incorporated a. Loss of ventilation/room heat-up calculation conclusions incorporated
09/99	Westinghouse Owner's Group (WOG) peer review completed
06/00	Incorporated loss of offsite power and offsite power restoration calculations
09/02	NUREG/CR-5750 used as source of general initiating event frequencies a. Incorporated some of the peer review level A and B findings and observations

Date	Model Change (continued)
2004	Added main feedwater and condensate systems to the secondary cooling function.
2005	MSPI (Mitigating Systems Performance Indicator) Model Update completed a. plant specific data b. reliability: 01/01/2000-12/31/2004 c. unavailability: January, 2002 to December, 2004 d. initiating events: 1990 to 12/31/2004 e. addressed remaining A and B level peer review findings and observations
2006	2005 Mod A Model (M305 Mod A) a. Revised the cooling dependency for the charging pump oil cooling system (CCE). SW is not required to cool charging pumps if auxiliary building temperatures remain below 90F.
2006	2005 Mod B and C Model (M305 Mod B & C) a. added internal flooding in Mod B b. revised junction box flood damage logic in internal flooding model in Mod C
2007	2005 Mod D Model (M305 Mod D) a. added hot leg recirculation to large loss of coolant accident (LOCA) b. added new pre-initiator human error probabilities (HEPs) c. updated Human Reliability Analysis (HRA) using latest methodology: Cause Based Decision Tree (CBDT), Human Cognitive Reliability Correlation (HCR), Technique for Human Error Rate Prediction (THERP) d. updated interfacing system LOCA e. updated Level 2 f. various other changes (e.g., replaced logic that assumed LOCA, steam generator tube rupture (SGTR) or steam line break (SLB) occurs in one reactor coolant system (RCS) loop or steam generator)
2008	Model updated to meet RG 1.200 (M308A)

Comprehensive Critical Reviews

The MPS3 PRA model has benefited from the following comprehensive technical PRA Peer Reviews:

NEI PRA Peer Review

The MPS3 internal events PRA received a formal industry PRA Peer Review in 1999 [Ref. 6]. The purpose of the PRA Peer Review process is 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 MPS3 review team used the "Westinghouse Owner's Group (WOG) Peer Review Process Guidance" as the basis for the review.

The general scope of the implementation of the PRA Peer Review included a 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 Large Early Release Frequency (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. All comments in Categories A and B have been addressed.

MPS3 PRA Self-Assessment

A self-assessment/independent review of the MPS3 PRA against the American Society of Mechanical Engineers (ASME) PRA standard was performed by Dominion with the support of a contracting company, MARACOR, in late 2007 using guidance provided in NRC RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results from Risk-Informed Activities" [Ref. 7]. This self-assessment was documented and used as a planning guide for the MPS3 2008 model update.

Many of the supporting requirements (SRs) identified in the self-assessment as not meeting Capability Category II have been incorporated into the MPS3 2008 model of record (M308A). Several improvements made to the model involved documenting sources of uncertainty/assumptions, including additional loss of single AC and DC buses initiators, upgrading component boundaries to be consistent with generic data, updating several thermal hydraulic (e.g., MAAP computer code) runs and improving success criteria documentation.

In the M308A model update, nearly all of the remaining SRs were addressed by further upgrades to the model documentation as well as improvements to the model. Of the 321 SRs, the MPS3 PRA does not meet 47 Category II SRs. Thirty-nine (39) of the 47 "not met" requirements pertain to various documentation issues or completion of the uncertainty analysis. There are eight "not met" SRs due to logic modeling, which fall into five categories; 1) Modeling HVAC dependencies for rooms/compartments without room heatup calculations, 2) review of inspection procedures for pre-initiator HEPs, 3) develop alignment-specific basic events based on plant OE, 4) include non-piping failures in the internal flooding analysis (i.e., expansion joints, bellows, and inadvertent sprinkler actuation), and 5) modeling RCS depressurization for SGTR events. Table 1 provides the status of identified gaps.

Table 1 Status of Identified Gaps to NEI 00-02 and Capability Category II of the ASME PRA Standard

Title	Description	NEI Element / ASME SR	Current Status / Comment	Importance to Application
Gap #1	Engineering analysis is required to screen support systems for accident initiation or mitigation.	IE-B3 SY-A19 SY-B6	The following HVAC dependencies are not modeled: • Emergency Diesel Generator (EDG) Sequencer HVAC • Normal Switchgear (not a significant impact) • Reactor Plant Component Cooling Water (RPCCW) Pumps	This is a model logic issue associated with the lack of modeling a couple HVAC dependencies. The PRA sensitivity study model will include logic modeling of the HVAC dependencies or an engineering analysis will be performed to screen out the aforementioned not-modeled HVAC dependencies.
Gap #2	Review of procedures and practices, those test and maintenance activities that require realignment of equipment outside its normal operational or standby status. The systematic review performed for identification of pre-initiator HEPs did not include a review of inspection procedures.	HR-A1	Review Inspection Procedures for Pre-Initiator HEPs.	This is a potential model logic issue. Therefore, inspection procedures will be reviewed to identify pre-initiator HEPs. Any identified pre-initiator HEP will be included in the PRA sensitivity study analysis.
Gap #3	Develop alignment-specific basic event values based on actual plant OE.	DA-C8	M308A meets Capability Category 1, in that the PRA assumes an overall average distribution of system alignments. The estimates used are reasonable. However, this approach does not meet Category 2 requirements.	This is a model logic issue. Therefore, the sensitivity study model will include alignment specific basic events values based on actual plant operating history.

Table 1 Status of Identified Gaps to NEI 00-02 and Capability Category II of the ASME PRA Standard				
Title	**Description**	**NEI Element / ASME SR**	**Current Status / Comment**	**Importance to Application**
Gap #4	Identify flooding sources that may result in fluid release due to failure of other (non-piping) components, human error during maintenance, or inadvertent sprinkler actuation.	IF-B2	Need to incorporate internal flooding frequencies associated with non-piping failures (e.g., expansion joints, bellows, and inadvertent sprinkler actuation).	This is a model logic issue. Therefore, the sensitivity study model will include non-piping flooding initiating event frequencies.
Gap #5	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.	LE-C2a LE-C6	The SAMGs have not been completely incorporated into the MPS3 Level 2 analysis. Need to credit SAMG operator action for depressurization of the RCS to allow low pressure injection to prevent induced steam generator tube rupture.	This is a conservative vulnerability in the model logic and, therefore, will be reviewed as part of the sensitivity study.

External Events Considerations

Internal Plant Examination – External Events (IPEEE)

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.

External Events

The MPS3 PRA is a Level 1 and 2 model that includes internal events and internal floods. For external events such as fire, seismic, extreme winds and other external events, the risk assessments from the IPEEE [Ref. 3] can be used for insights on changes to surveillance intervals.

Fire Risk

The MPS3 PRA does not include a fire model. Therefore, the results of the external events fire risk assessment performed for the IPE can be qualitatively assessed for insights on changes to surveillance intervals. The IPE fire risk analysis quantified a core damage frequency (CDF) impact by combining the frequency of fires and the probability of detection/suppression failure with the remaining safety function unavailabilities. A systematic approach was used to identify critical fire areas where fires could fail safety functions and pose an increased risk of core damage if other safety functions are unavailable. The CDF due to fires is 4.8E-06/yr, with the dominant risk being fires in the cable spreading room, switchgear rooms, control room, and auxiliary building.

Seismic Risk

The MPS3 PRA has not updated the seismic model since the IPE. Therefore, the results of the external events seismic risk assessment performed for the IPE were reviewed to qualitatively assess the impact of the surveillance test frequency extension on seismic event risk. The IPE seismic risk analysis quantified a CDF impact by combining the seismic hazard frequencies with the fragilities of critical structures and components and the safety function unavailabilities to obtain a CDF. The CDF due to seismic events is 9.1E-06/yr, with the dominant risk being seismic events that result in a loss of offsite power and failure of the EDG enclosures, or collapse of the control building.

High Winds, Floods and Other External Events

The risk of other external events such as high winds, aircraft accidents, hazardous materials and turbine missiles was assessed in the MPS3 IPEEE. The IPEEE assessments concluded that the risk of these accidents is negligible primarily due to the low frequency of occurrence that would cause damage to mitigating systems. For example, reinforced concrete houses provide the applicable safety systems missile protection during high wind conditions.

Summary of External Event Status

As stated earlier, the NEI 04-10, Revision 1 methodology allows for STI change evaluations to be performed in the absence of quantifiable PRA models for external hazards. Therefore, in performing the assessments for the other hazard groups, a qualitative or bounding approach will be utilized in most cases. This approach is consistent with the accepted NEI 04-10, Revision 1 methodology.

SUMMARY

The MPS3 PRA technical capability evaluations and the maintenance and update processes described above provide a robust basis for concluding that the full power internal events MPS3 PRA is suitable for use in risk-informed processes such as that proposed for the implementation of a Surveillance Frequency Control Program. 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.

REFERENCES

1. Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies, Industry Guidance Document, NEI 04-10, Revision 1, April 2007.
2. Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 1, January 2007.
3. E. J. Mroczka letter to the Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Response to Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities, Summary Report Submittal," dated August 31, 1990.
4. V. L. Rooney (NRC) letter to Northeast Nuclear Energy Company, "Staff Evaluation of Millstone 3 Individual Plant Examination, (IPE) -Internal Events, GL 88-20 (TAC No. M74434)," May 5, 1992.
5. J. W. Andersen (NRC) letter to Northeast Nuclear Energy Company, "Millstone Nuclear Power Station, Unit No. 3 Individual Plant Examination of External Events (TAC No. M83643)," May 26, 1998.

6. Millstone Power Station Unit 3 Probabilistic Risk Assessment Peer Review Report, September 1999
7. Millstone Power Station Unit 3 Probabilistic Risk Assessment Model Notebook Part IV, Appendix A.1, "Internal Events Model Self Assessment," August 2009

ATTACHMENT 3

MARKED-UP TECHNICAL SPECIFICATIONS CHANGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the SHUTDOWN MARGIN not within the limits specified in the COLR, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be within the limits specified in the COLR:

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

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

the frequency specified in the Surveillance Frequency Control Program

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

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

* See Special Test Exceptions Specification 3.10.1.

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

the frequency specified in the Surveillance Frequency Control Program

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 3, 4 AND 5 LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.1.1.1.2 The SHUTDOWN MARGIN shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).*

APPLICABILITY: MODES 3, 4 and 5

ACTION:

With the SHUTDOWN MARGIN less than the required value, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be within the limits specified in the COLR:

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

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

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

4.1.1.1.2.2 Valve 3CHS*V305 shall be verified closed and locked at ~~least once per 31 days~~.

* Additional SHUTDOWN MARGIN requirements, if required, are given in Specification 3.3.5.

the frequency specified in the Surveillance Frequency Control Program

SHUTDOWN MARGIN - COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to

a. the limits specified in the CORE OPERATING LIMITS REPORT (COLR) for MODE 5 with RCS loops not filled* or

b. the limits specified in the COLR for MODE 5 with RCS loops filled* with the chemical and volume control system (CVCS) aligned to preclude reactor coolant system boron concentration reduction.

APPLICABILITY: MODE 5 LOOPS NOT FILLED

ACTION:

a. With the SHUTDOWN MARGIN less than the above, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

b. With the CVCS dilution flow paths not closed and secured in position in accordance with Specification 3.1.1.2(b), immediately close and secure the paths or meet the limits specified in the COLR for MODE 5 with RCS loops not filled.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be within the limits specified in the COLR:

the frequency specified in the Surveillance Frequency Control Program

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

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

1. Reactor Coolant System boron concentration,

2. Control rod position,

3. Reactor Coolant System average temperature,

4. Fuel burnup based on gross thermal energy generation,

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* Additional SHUTDOWN MARGIN requirements, if required, are given in Specification 3.3.5.

SURVEILLANCE REQUIREMENTS (Continued)

the frequency specified in the Surveillance Frequency Control Program

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.2.2 At ~~least once per 31 days~~ the following valves shall be verified closed and locked. The valves may be opened on an intermittent basis under administrative controls except as noted.

Valve Number		Valve Function	Valve Position
1.	V304(Z-)	Primary Grade Water to CVCS	Closed
2.	V120(Z-)	Moderating Hx Outlet	Closed
3.	V147(Z-)	BTRS Outlet	Closed
4.	V797(Z-)	Failed Fuel Monitoring Flushing	Closed
5.	V100(Z-)	Resin Sluice, CVCS Cation Bed Demineralizer	Closed
6.	V571(Z-)	Resin Sluice, CVCS Cation Bed Demineralizer	Closed
7.	V111(Z-)	Resin Sluice, CVCS Cation Bed Demineralizer	Closed
8.	V112(Z-)	Resin Sluice, CVCS Cation Bed Demineralizer	Closed
9.	V98(Z-)/V99(Z-)	Resin Sluice, CVCS Mixed Bed Demineralizer	Closed
10.	V569(Z-)/V570(Z-)	Resin Sluice, CVCS Mixed Bed Demineralizer	Closed
11.	V107(Z-)/V109(Z-)	Resin Sluice, CVCS Mixed Bed Demineralizer	Closed
12.	V108(Z-)/V110(Z-)	Resin Sluice, CVCS Mixed Bed Demineralizer	Closed
13.	V305(Z-)*	Primary Grade Water to Charging Pumps	Closed

* This valve may not be opened under administrative controls.

ACTION:(Continued)

c. A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d. THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a. above, POWER OPERATION may continue provided that:

1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ±12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

d. With more than one rod misaligned from its group step counter demand height by more than ±12 steps (indicated position), be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at ~~least once per 92 days~~.

the frequency specified in the Surveillance Frequency Control Program

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ±12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

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

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

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

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

the frequency specified in the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

4.1.3.2.1 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at ~~least once per 12 hours~~ except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

4.1.3.2.2 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel at ~~least once per 24 months.~~

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

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

a. T_{avg} greater than or equal to 500°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 65% of RATED THERMAL POWER with the reactor coolant stop valves in the nonoperating loop closed.

SURVEILLANCE REQUIREMENTS

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

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

b. Deleted

c. At ~~least once per 24 months.~~

the frequency specified in the Surveillance Frequency Control Program

SHUTDOWN ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limits specified in the COLR except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

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

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limits specified in the COLR:

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

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

the frequency specified in the Surveillance Frequency Control Program

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* See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

** With K_{eff} greater than or equal to 1.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With the control banks inserted beyond the insertion limits specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

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

SURVEILLANCE REQUIREMENTS

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

the frequency specified in the Surveillance Frequency Control Program

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* See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

** With K_{eff} greater than or equal to 1.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

4.2.1.1.4 When in base load operation, the target flux difference shall be updated at ~~least once per 31 Effective Full Power Days~~ by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.1.3 or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

the frequency specified in the Surveillance Frequency Control Program

c. Satisfying the following relationship:

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

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

where $F_Q{}^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, $F_Q{}^{RTP}$ is the F_Q limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the relative THERMAL POWER, and W(Z) is the cycle-dependent function that accounts for power distribution transients encountered during normal operation. $F_Q{}^{RTP}$, K(Z), and W(Z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.6.

d. Measuring $F_Q{}^M(Z)$ according to the following schedule:

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

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

the frequency specified in the Surveillance Frequency Control Program

e. With the maximum value of

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

over the core height (Z) increasing since the previous determination of $F_Q{}^M(Z)$, either of the following ACTIONS shall be taken:

(1) Increase $F_Q{}^M(Z)$ by an appropriate factor specified in the COLR and verify that this value satisfies the relationship in Specification 4.2.2.1.2.c, or

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

b. During base load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.1.3.a shall be satisfied before reentering base load operation.

4.2.2.1.4 During base load operation $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

b. Evaluate the computed heat flux hot channel factor by performing both of the following:

(1) Determine the computed heat flux hot channel factor, $F_Q^M(Z)$, by increasing the measured $F_Q^M(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties, and

(2) Verify that $F_Q^M(Z)$ satisfies the requirements of Specification 3.2.2.1 for all core plane regions, i.e., 0 - 100% inclusive.

c. Satisfying the following relationship:

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

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the relative THERMAL POWER, and $W(Z)_{BL}$ is the cycle-dependent function that accounts for limited power distribution transients encountered during base load operation. F_Q^{RTP}, K(Z), and $W(Z)_{BL}$ are specified in the COLR as per Specification 6.9.1.6.

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

(1) Prior to entering base load operation after satisfying Section 4.2.2.1.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative THERMAL POWER having been maintained above APL^{ND} for the 24 hours prior to mapping, and

(2) At ~~least once per 31 Effective Full Power Days.~~ the frequency specified in the Surveillance Frequency Control Program

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate that the RCS total flow rate is restored to within the limits specified above and in the COLR and $F^N_{\Delta H}$ is restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.1.2 $F^N_{\Delta H}$ shall be determined to be within the acceptable range:

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

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

4.2.3.1.3 The RCS total flow rate shall be determined to be within the acceptable range by:

a. Verifying by precision heat balance that the RCS total flow rate is ≥ 363,200 gpm and greater than or equal to the limit specified in the COLR within 24 hours after reaching 90% of RATED THERMAL POWER after each fuel loading, and

the frequency specified in the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (Continued)

b. Verifying that the RCS total flow rate is ≥ 363,200 gpm and greater than or equal to the limit specified in the COLR at ~~least once per 12 hours~~.

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

4.2.3.1.5 DELETED.

4.2.3.1.6 DELETED.

the frequency specified in the Surveillance Frequency Control Program

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at ~~least once per 12 hours~~.

the frequency specified in the Surveillance Frequency Control Program

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR):

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the above DNB-related parameters shall be verified to be within the limits specified in the COLR at ~~least once per 12 hours~~.

the frequency specified in the Surveillance Frequency Control Program

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-1.

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements ~~specified in Table 4.3-1~~.

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

the frequency specified in the Surveillance Frequency Control Program

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

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	~~R~~(14)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	~~S~~	~~D~~(2, 4), ~~M~~(3, 4), ~~Q~~(4, 6), ~~R~~(4, 5)	~~Q~~	N.A.	N.A.	1, 2
b. Low Setpoint	~~S~~	~~R~~(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	~~R~~(4, 5)	~~Q~~	N.A.	N.A.	1, 2
4. Deleted						
5. Intermediate Range	~~S~~	~~R~~(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	~~S~~	~~R~~(4, 5)	S/U(1), ~~Q~~(9)	N.A.	N.A.	2**, 3*, 4*, 5*
7. Overtemperature ΔT	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	1, 2
8. Overpower ΔT	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	~~S~~	~~R~~	~~Q~~(18)	N.A.	N.A.	1*****
10. Pressurizer Pressure--High	~~S~~	~~R~~	~~Q~~(18)	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	1*****
12. Reactor Coolant Flow--Low	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	1

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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low	~~S~~	~~R~~	~~Q~~(18)	N.A.	N.A.	1, 2
14. Low Shaft Speed - Reactor Coolant Pumps	N.A.	~~R~~(13)	~~Q~~	N.A.	N.A.	1
15. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	~~R~~	N.A.	S/U(1, 10)****	N.A.	1
b. Turbine Stop Valve Closure	N.A.	~~R~~	N.A.	S/U(1, 10)****	N.A.	1
16. Deleted						
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	~~R~~(4)	~~R~~	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	~~R~~(4)	~~R~~	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	~~R~~(4)	~~R~~	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	~~R~~(4)	~~R~~	N.A.	N.A.	1
e. Power Range Neutron Flux, P-10	N.A.	~~R~~(4)	~~R~~	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	~~R~~	~~R~~	N.A.	N.A.	1

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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
18. Reactor Trip Breaker	N.A.	N.A.	N.A.	~~M~~(~~7,~~ 11)	N.A.	1, 2, 3*, 4*, 5*
19. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	~~M(7)~~	1, 2, 3*, 4*, 5*
20. DELETED						
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	~~M~~(~~7,~~ 15) ~~R~~(16)	N.A.	1, 2, 3*, 4*, 5*
22. DELETED						

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*	When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.
**	Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
***	Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.
****	Above the P-9 (Reactor Trip/Turbine Interlock) Setpoint.
*****	Above the P-7 (At Power) Setpoint
(1)	If not performed in previous 31 days.
(2)	Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.
(3)	Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(4)	Neutron detectors may be excluded from CHANNEL CALIBRATION.
(5)	Detector plateau curves shall be obtained, and evaluated and compared to manufacturer's data. For the Source Range, Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(6)	Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(7)	~~Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.~~ Deleted
(8)	(Not used)
(9)	~~Quarterly~~ surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P–6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-2.

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements ~~specified in Table 4.3-2.~~

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

the frequency specified in the Surveillance Frequency Control Program

* The provisions of Specification 4.0.4 are not applicable for response time verification of steam line isolation for entry into MODE 4 and MODE 3 and turbine driven auxiliary feedwater pump for entry into MODE 3.

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

TABLE 4.3-2
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~M(1)~~	~~M(1)~~	~~Q~~(4)	1, 2, 3, 4
c. Containment Pressure-High-1	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~M(1)~~	~~M(1)~~	~~Q~~(4)	1, 2, 3, 4
c. Containment Pressure-High-3	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1. Manual Initiation	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3, 4
2. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~M(1)~~	~~M(1)~~	~~Q~~ (4)	1, 2, 3, 4
3. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1. Manual Initiation	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3, 4
2. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~M(1)~~	~~M(1)~~	~~Q~~ (4)	1, 2, 3, 4
3. Containment Pressure-High-3	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
c. DELETED								
4. Steam Line Isolation								
d. Manual Initiation								
1. Individual	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3, 4
2. System	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3, 4

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

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation (Continued)								
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~M(1)~~	~~M(1)~~	~~Q~~(4)	1, 2, 3, 4
c. Containment Pressure-High-2	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Steam Line Pressure-Low	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~M(1)~~	~~M(1)~~	~~Q~~(4)	1, 2
b. Steam Generator Water Level-High-High	~~S~~	~~R~~	~~Q~~	N.A.	~~M(1)~~	~~M(1)~~	~~Q~~(4)	1, 2, 3
c. Safety Injection Actuation Logic	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2
d. T_{ave} Low Coincident with Reactor Trip (P-4)	N.A.	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~M(1)~~	~~M(1)~~	~~Q~~(4)	1, 2, 3
c. Steam Generator Water Level-Low-Low	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power	See Item 8. below for all Loss of Power Surveillance.							
f. Containment Depressurization Actuation (CDA)	See Item 2. above for all CDA Surveillance Requirements.							
7. Control Building Isolation								
a. Manual Actuation	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	*
b. Manual Safety Injection Actuation	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~M(1)~~	~~M(1)~~	~~Q~~(4)	1, 2, 3, 4
d. Containment Pressure--High-1	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Control Building Isolation (Continued)								
e. Control Building Inlet Ventilation Radiation	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	*
8. Loss of Power								
a. 4 kV Bus Undervoltage (Loss of Voltage)	N.A.	~~R~~	N.A.	~~M~~(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4 kV Bus Undervoltage (Grid Degraded Voltage)	N.A.	~~R~~	N.A.	~~M~~(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	~~R~~	N.A.	N.A.	N.A.	1, 2, 3
10. Emergency Generator Load Sequencer	N.A.	N.A.	N.A.	N.A.	~~Q~~(~~1~~, 2)	N.A.	N.A.	1, 2, 3, 4
11. Cold Leg Injection Permissive, P-19	~~S~~	~~R~~	~~Q~~	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

Deleted

1. ~~Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.~~

2. This surveillance may be performed continuously by the emergency generator load sequencer auto test system as long as the EGLS auto test system is demonstrated OPERABLE by the performance of an ACTUATION LOGIC TEST at ~~least once per 92 days.~~ At the frequency specified in the Surveillance Frequency Control Program

3. ~~On a monthly basis,~~ a loss of voltage condition will be initiated at each undervoltage monitoring relay to verify individual relay operation. Setpoint verification and actuation of the associated logic and alarm relays will be performed as part of the CHANNEL CALIBRATION ~~required once per 18 months~~.

4. For Engineered Safety Features Actuation System functional units with only Potter & Brumfield MDR series relays used in a clean, environmentally controlled cabinet, as discussed in Westinghouse Owners Group Report WCAP- 13900, the surveillance interval for slave relay testing is ~~R~~ the frequency specified in the Surveillance Frequency Control Program.

* MODES 1, 2, 3, and 4.
During movement of recently irradiated fuel assemblies.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations 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 for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

required

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST for the MODES and at the frequencies ~~shown in Table 4.3-3~~.

specified in the Surveillance Frequency Control Program

Delete Table 4.3-3

~~**TABLE 4.3-3**~~

~~**RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS**~~

~~FUNCTIONAL UNIT~~	~~CHANNEL CHECK~~	~~CHANNEL CALIBRATION~~	~~ANALOG CHANNEL OPERATIONAL TEST~~	~~MODES FOR WHICH SURVEILLANCE IS REQUIRED~~
~~1. Containment~~				
~~a. Deleted~~				
~~b. RCS Leakage Detection~~				
~~1) Particulate Radio-activity~~	~~S~~	~~R~~	~~Q~~	~~1, 2, 3, 4~~
~~2) Deleted~~				
~~2. Fuel Storage Pool Area Monitors~~				
~~a. Radiation Level~~	~~S~~	~~R~~	~~Q~~	~~*~~

~~TABLE NOTATIONS~~

* ~~With fuel in the fuel storage pool area.~~

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown Instrumentation transfer switches, power, controls and monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With one or more Remote Shutdown Instrumentation transfer switches, power, or control circuits inoperable, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

c. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

SURVEILLANCE REQUIREMENTS

required

4.3.3.5.1 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~~.

specified in the Surveillance Frequency Control Program

4.3.3.5.2 Each Remote Shutdown Instrumentation transfer switch, power and control circuit including the actuated components, shall be demonstrated OPERABLE at ~~least once per 18 months~~.

the frequency specified in the Surveillance Frequency Control Program

Delete Table 4.3-6

TABLE 4.3-6
REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Reactor Trip Breaker Indication	M	N.A.
2.	Pressurizer Pressure	M	R
3.	Pressurizer Level	M	R
4.	Steam Generator Pressure	M	R
5.	Steam Generator Water Level	M	R
6.	Auxiliary Feedwater Flow Rate	M	R
7.	Loop Hot Leg Temperature	M	R
8.	Loop Cold Leg Temperature	M	R
9.	Reactor Coolant System Pressure (Wide Range)	M	R
10.	DWST Level	M	R
11.	RWST Level	M	R
12.	Containment Pressure	M	R
13.	Emergency Bus Voltmeters	M	R
14.	Source Range Count Rate	M*	R
15.	Intermediate Range Amps	M	R
16.	Boric Acid Tank Level	M	R

* When below P-6 (intermediate range neutron flux interlock setpoint).

action taken, the cause of the inoperability, and the plans and schedule for restoring the channel to OPERABLE status.

f. With the number of OPERABLE channels for the reactor vessel water level monitor less than the minimum channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory;

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the channel(s) to OPERABLE status; and

3. Restore the channel(s) to OPERABLE status at the next scheduled refueling.

g. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

SURVEILLANCE REQUIREMENTS

required

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

4.3.3.6.2 Deleted

specified in the Surveillance Frequency Control Program

Delete Table 4.3-7

TABLE 4.3-7
ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure		
a. Normal Range	M	R
b. Extended Range	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Demineralized Water Storage Tank Water Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Reactor Coolant System Subcooling Margin Monitor	M	R
13. Containment Water Level (Wide Range)	M	R
14. Core Exit Thermocouples	M	R
15. DELETED		

Delete Table 4.3-7

TABLE 4.3-7 (Continued)
ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
16.	Containment Area - High Range Radiation Monitor	M	R*
17.	Reactor Vessel Water Level	M	R**
18.	Deleted		
19.	Neutron Flux	M	R

* CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

** Electronic calibration from the ICC cabinets only.

SURVEILLANCE REQUIREMENTS

4.3.5 a. Each of the above required shutdown margin monitoring instruments shall be demonstrated OPERABLE by an ANALOG CHANNEL OPERATIONAL TEST at ~~least once per 92 days~~ that shall include verification that the Shutdown Margin Monitor is set per the CORE OPERATING LIMITS REPORT (COLR).

b. At ~~least once per 24 hours~~ VERIFY the minimum count rate (counts/sec) as defined within the COLR.

the frequency specified in the Surveillance Frequency Control Program

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

the frequency specified in the Surveillance Frequency Control Program

*See Special Test Exceptions Specification 3.10.4.

SURVEILLANCE REQUIREMENTS

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

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 17% ~~at least once per 12 hours~~.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant ~~at least once per 12 hours~~.

at the frequency specified in the Surveillance Frequency Control Program

LIMITING CONDITION FOR OPERATION (continued)

b. With less than the above required reactor coolant loops in operation and the Control Rod Drive System is capable of rod withdrawal, within 1 hour open the Reactor Trip System breakers.

c. With no loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1.2 and immediately initiate corrective action to return the required loop to operation.

SURVEILLANCE REQUIREMENTS

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

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 17% ~~at least once per 12 hours~~.

4.4.1.3.3 The required loop(s) shall be verified in operation and circulating reactor coolant ~~at least once per 12 hours~~.

at the frequency specified in the Surveillance Frequency Control Program

ACTION:

a. With less than the required RHR loop(s) OPERABLE or with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1.1.2 and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

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

at the frequency specified in the Surveillance Frequency Control Program

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

SURVEILLANCE REQUIREMENTS

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

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

at the frequency specified in the Surveillance Frequency Control Program

REACTOR COOLANT SYSTEM

LOOP STOP VALVES

LIMITING CONDITION FOR OPERATION

3.4.1.5 Each RCS loop stop valve shall be open and the power removed from the valve operator.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With power available to one or more loop stop valve operators, remove power from the loop stop valve operators within 30 minutes or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b.*(1) With one or more RCS loop stop valves closed, maintain the valve(s) closed and be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.5 Verify each RCS loop stop valve is open and the power removed from the valve operator at ~~least once per 31 days~~.

the frequency specified in the Surveillance Frequency Control Program

*(1) All required ACTIONS of ACTION Statement 3.4.1.5.b shall be completed whenever this action is entered.

3/4.4.3 PRESSURIZER

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.3.1 The pressurizer shall be OPERABLE with:

a. at least two groups of pressurizer heaters, each having a capacity of at least 175 kW; and

b. water level maintained at programmed level +/-6% of full scale (Figure 3.4-5).

APPLICABILITY: MODES 1 and 2.

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.

b. With pressurizer water level outside the parameters described in Figure 3.4-5, within 2 hours restore programmed level to within +/- 6% of full scale, or be in at least HOT STANDBY within the next 6 hours.

c. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1.1 The pressurizer water level shall be verified to be within programmed level +/- 6% of full scale at ~~least once per 12 hours~~.

4.4.3.1.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at ~~least once each refueling interval~~.

the frequency specified in the Surveillance Frequency Control Program

LIMITING CONDITION FOR OPERATION

3.4.3.2 The pressurizer shall be OPERABLE with:

a. at least two groups of pressurizer heaters, each having a capacity of at least 175 kW; and

b. water level less than or equal to 89% of full scale.

APPLICABILITY: MODE 3

ACTION:

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

b. With the pressurizer otherwise inoperable, be in HOT SHUTDOWN within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The pressurizer water level shall be determined to be less than or equal to 89% of full scale at ~~least once per 12 hours~~.

4.4.3.2.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at ~~least once each refueling interval~~.

the frequency specified in the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

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

a. Performance of a CHANNEL CALIBRATION at ~~least once per 24 months~~; and

b. Operating the valve through one complete cycle of full travel during MODES 3 or 4 at ~~least once per 24 months~~; and

c. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV high pressurizer pressure actuation channels, but excluding valve operation, at ~~least once each quarter~~; and

d. Verify the PORV high pressure automatic opening function is enabled at ~~least once per 12 hours~~.

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

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

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Particulate Radioactivity Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in ~~Table 4.3-3~~, and

b. Containment Drain Sump Monitoring System-performance of CHANNEL CALIBRATION at ~~least once per 24 months~~.

the Surveillance Frequency Control Program

the frequency specified in the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

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

a. Deleted

b. Deleted

the frequency specified in the Surveillance Frequency Control Program

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2250 ± 20 psia at ~~least once per 31 days~~ with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

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

1. Not required to be performed until 12 hours after establishment of steady state operation.
2. Not applicable to primary to secondary LEAKAGE.

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the frequency specified in the Surveillance Frequency Control Program

d. Performance of a Reactor Coolant System water inventory balance at ~~least once per 72 hours~~;

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

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

----

e. Verification that primary to secondary LEAKAGE is ≤ 150 gallons per day through any one Steam Generator at ~~least once per 72 hours~~, and;

f. Monitoring the Reactor Head Flange Leakoff System at ~~least once per 24 hours~~.

4.4.6.2.2[(1)(2)]Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying LEAKAGE to be within its limit:

---

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

(2) This surveillance is not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the RHR flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

SURVEILLANCE REQUIREMENTS (Continued)

the frequency specified in the Surveillance Frequency Control Program

a. At ~~least once per 24 months~~,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Deleted

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve, and

e. When tested pursuant to Specification 4.0.5.

SURVEILLANCE REQUIREMENTS

4.4.8.1 Verify the specific activity of the reactor coolant less than or equal to 81.2 microCuries per gram DOSE EQUIVALENT XE-133 ~~once per 7 days~~.*

4.4.8.2 Verify the specific activity of the reactor coolant less than or equal to 1.0 microCuries per gram DOSE EQUIVALENT I-131 ~~once per 14 days~~,* and between 2 and 6 hours after a THERMAL POWER change of greater than or equal to 15% RATED THERMAL POWER within a one hour period.

at the frequency specified in the Surveillance Frequency Control Program

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

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Demonstrate that each required PORV is OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL 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 24 months~~; and

c. Verifying the PORV block valve is open and the PORV Cold Overpressure Protection System (COPPS) is armed at ~~least once per 72 hours~~ when the PORV is being used for overpressure protection.

4.4.9.3.2 Demonstrate that each required RHR suction relief valve is OPERABLE by:

a. Verifying the isolation valves between the RCS and each required RHR suction relief valve are open at ~~least once per 12 hours~~; and

b. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 When complying with 3.4.9.3.4, verify that the RCS is vented through a vent pathway ≥ 2.0 square inches at ~~least once per 31 days~~ for a passive vent path and at ~~least once per 12 hours~~ for unlocked open vent valves.

4.4.9.3.4 Verify that no Safety Injection pumps are capable of injecting into the RCS ~~at least once per 12 hours~~.

4.4.9.3.5 Verify that a maximum of one centrifugal charging pump is capable of injecting into the RCS at ~~least once per 12 hours~~.

the frequency specified in the Surveillance Frequency Control Program

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6618 and 7030 gallons,

c. A boron concentration of between 2600 and 2900 ppm, and

d. A nitrogen cover-pressure of between 636 and 694 psia.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

the frequency specified in the Surveillance Frequency Control Program

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At ~~least once per 12 hours~~ by:

1) Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks are within their limits, and

2) Verifying that each accumulator isolation valve is open.

b. At ~~least once per 31 days~~ and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution. This surveillance is not required when the volume increase makeup source is the RWST.

* Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

c. At ~~least once per 31 days~~ the frequency specified in the Surveillance Frequency Control Program when the RCS pressure is above 1000 psig by verifying that the associated circuit breakers are locked in a deenergized position or removed.

the frequency specified in the Surveillance Frequency Control Program

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

Valve Number	Valve Function	Valve Position
3SIH*MV8806	RWST Supply to SI Pumps	OPEN
3SIH*MV8802A	SI Pump A to Hot Leg Injection	CLOSED
3SIH*MV8802B	SI Pump B to Hot Leg Injection	CLOSED
3SIH*MV8835	SI Cold Leg Master Isolation	OPEN
3SIH*MV8813	SI Pump Master Miniflow Isolation	OPEN
3SIL*MV8840	RHR to Hot Leg Injection	CLOSED
3SIL*MV8809A	RHR Pump A to Cold Leg Injection	OPEN
3SIL*MV8809B	RHR Pump B to Cold Leg Injection	OPEN

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

1) Verifying that the ECCS piping, except for the operating centrifugal charging pump(s) and associated piping, the RSS pump, the RSS heat exchanger and associated piping, is full of water, and

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

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

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

2) At least once daily of the areas affected (during each day) within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

d. At ~~least once per 24 months~~ by:

1) Verifying automatic interlock action of the RHR System from the Reactor Coolant System by ensuring that with a simulated signal greater than or equal to 412.5 psia the interlocks prevent the valves from being opened.

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

e. At ~~least once per 24 months~~ by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection actuation test signal, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) the frequency specified in the Surveillance Frequency Control Program

c) RHR pump.

3) Verifying that the Residual Heat Removal pumps stop automatically upon receipt of a Low-Low RWST Level test signal.

f. By verifying that each of the following pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump

2) Safety Injection pump

3) RHR pump

4) Containment recirculation pump

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

1) Within 4 hours following completion of each valve stroking operation when the ECCS subsystems are required to be OPERABLE, and

2) At ~~least once per 24 months~~.

ECCS Throttle Valves

Valve Number	Valve Number
3SIH*V6	3SIH*V25
3SIH*V7	3SIH*V27

LIMITING CONDITION FOR OPERATION

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

a. A contained borated water volume between 1,166,000 and 1,207,000 gallons,

b. A boron concentration between 2700 and 2900 ppm of boron,

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

d. A maximum solution temperature of 50°F.

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

ACTION

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

SURVEILLANCE REQUIREMENTS

the frequency specified in the Surveillance Frequency Control Program

4.5.4 The RWST shall be demonstrated OPERABLE:

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

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

2) Verifying the boron concentration of the water.

b. At ~~least once per 24 hours~~ by verifying the RWST temperature.

3/4.5.5 pH TRISODIUM PHOSPHATE STORAGE BASKETS

LIMITING CONDITION FOR OPERATION

3.5.5 The trisodium phosphate (TSP) dodecahydrate Storage Baskets shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With the TSP Storage Baskets inoperable, restore the system TSP Storage Baskets to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The TSP Storage Baskets shall be demonstrated OPERABLE at ~~least once per 24 months~~ by verifying that a minimum total of 974 cubic feet of TSP is contained in the TSP Storage Baskets.

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 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

the frequency specified in the Surveillance Frequency Control Program

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; and

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

c. Deleted

---

(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 during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

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

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

LIMITING CONDITION FOR OPERATION

Continued

c. With the containment air lock inoperable, except as specified in ACTION a. or ACTION b. above, immediately initiate action to evaluate overall containment leakage rate per Specification 3.6.1.2 and verify an air lock door is closed within 1 hour. Restore the air lock to OPERABLE status within 24 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying leakage results 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).

b. Deleted

c. At ~~least once per 24 months~~ by verifying that only one door in each air lock can be opened at a time.

the frequency specified in the Surveillance Frequency Control Program

CONTAINMENT SYSTEMS

CONTAINMENT PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment pressure shall be maintained greater than or equal to 10.6 psia and less than or equal to 14.0 psia.

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

ACTION:

With the containment pressure less than 10.6 psia or greater than 14.0 psia, restore the containment pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

the frequency specified in the Surveillance Frequency Control Program

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained greater than or equal to 80°F and less than or equal to 120°F.

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

ACTION:

With the containment average air temperature less than 80°F or greater than 120°F, restore the average air temperature to within the limit within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

the frequency specified in the Surveillance Frequency Control Program

Location

a. 94 ft elevation, E outside crane wall

b. 86 ft elevation, NW outside crane wall

c. 75 ft elevation, W Steam Generator platform

d. 75 ft elevation, E Steam Generator platform

e. 45 ft elevation, Pressurizer cubicle, crane wall

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and each 42-inch containment shutdown purge supply and exhaust isolation valve shall be closed and locked closed.

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

ACTION:

a. With a 42-inch containment purge supply and/or exhaust isolation valve open or not locked closed, close and/or lock close that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 The containment purge supply and exhaust isolation valves shall be verified to be locked closed and closed at ~~least once per 31 days~~.

the frequency specified in the Surveillance Frequency Control Program

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Quench Spray subsystems shall be OPERABLE.

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

ACTION:

With one Containment Quench Spray subsystem inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Quench Spray subsystem shall be demonstrated OPERABLE:

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

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

2) Verifying the temperature of the borated water in the refueling water storage tank is between 40°F and 50°F.

b. By verifying that each pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5;

c. At ~~least once per 24 months~~, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal, and

2) Verifying that each spray pump starts automatically on a CDA test signal.

d. By verifying each spray nozzle is unobstructed following maintenance that could cause nozzle blockage.

the frequency specified in the Surveillance Frequency Control Program

LIMITING CONDITION FOR OPERATION

3.6.2.2 Two independent Recirculation Spray Systems shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each Recirculation Spray System shall be demonstrated OPERABLE:

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

b. By verifying that each pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5;

c. At ~~least once per 24 months~~ by verifying that on a CDA test signal, each recirculation spray pump starts automatically after receipt of an RWST Low-Low signal;

d. At ~~least once per 24 months~~, by verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal; and

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

the frequency specified in the Surveillance Frequency Control Program

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE.[(1)] [(2)]

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

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation barrier OPERABLE in the affected penetration(s), and:

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 deactivated automatic valve(s) secured in the isolation position(s), or

c. Isolate the affected penetration(s) within 4 hours by use of 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 at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 DELETED

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at ~~least once per 24 months~~ the frequency specified in the Surveillance Frequency Control Program by:

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

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

c. Verifying that on a Containment High Radiation test signal, each purge supply and exhaust isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

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(1) The provisions of this Specification 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.

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

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

STEAM JET AIR EJECTOR

LIMITING CONDITION FOR OPERATION

3.6.5.1 The inside and outside isolation valves in the steam jet air ejector suction line shall be closed.

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

ACTION:

With the inside or outside isolation valves in the steam jet air ejector suction line not closed, restore the valve to the closed position within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1.1 The steam jet air ejector suction line outside isolation valve shall be determined to be in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 200°F and at ~~least once per 31 days thereafter~~.

4.6.5.1.2 The steam jet air ejector suction line inside isolation valve shall be determined to be locked in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 200°F.

the frequency specified in the Surveillance Frequency Control Program

3/4.6.6 SECONDARY CONTAINMENT

SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Supplementary Leak Collection and Release Systems shall be OPERABLE with each system comprised of:

a. one OPERABLE filter and fan, and

b. one OPERABLE Auxiliary Building Filter System as defined in Specification 3.7.9.

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

ACTION:

With one Supplementary Leak Collection and Release System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

the frequency specified in the Surveillance Frequency Control Program

4.6.6.1 Each Supplementary Leak Collection and Release System 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 filters and charcoal adsorbers and verifying a system flow rate of 7600 cfm to 9800 cfm and that the system operates for at least 10 continuous hours with the heaters operating.

b. At ~~least once per 24 months~~ or following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978,* and the system flow rate is 7600 cfm to 9800 cfm;

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,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F) and a relative humidity of 70%; and

3) Verifying a system flow rate of 7600 cfm to 9800 cfm during system operation when tested in accordance with ANSI N510-1980.

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,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F) and a relative humidity of 70%:

the frequency specified in the Surveillance Frequency Control Program

d. At ~~least once per 24 months~~ by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.25 inches Water Gauge while operating the system at a flow rate of 7600 cfm to 9800 cfm,

2) Verifying that the system starts on a Safety Injection test signal, and

3) Verifying that the heaters dissipate 50 ±5 kW when tested in accordance with ANSI N510-1980.

* ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

SECONDARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.6.6.2 Secondary Containment shall be OPERABLE.

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

ACTION:

With Secondary Containment inoperable, restore Secondary Containment to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENT

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

4.6.6.2.2 At ~~least once per 24 months~~, verify each Supplementary Leak Collection and Release System produces a negative pressure of greater than or equal to 0.4 inch water gauge in the Auxiliary Building at 24'-6" elevation within 120 seconds after a start signal.

the frequency specified in the Surveillance Frequency Control Program