Dominion Nuclear Connecticut, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060
Web Address: www.dom.com

Dominion®

October 4, 2012

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

Serial No. 12-580
NSSL/MLC R0
Docket No. 50-423
License No. NPF-49

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

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.

Attachment 1 provides a description and assessment of the proposed change. Attachment 2 includes DNC documentation with regard to the technical adequacy of the Probabilistic Risk Assessment. Attachment 4 provides a cross-reference between the NUREG-1431 surveillances included in TSTF-425 versus the MPS3 surveillances included in this amendment request. Attachments 3 and 6 provide the MPS3 marked-up TS pages and TS Bases pages, respectively. The marked-up TS Bases pages are provided for information only. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program upon approval of this amendment request.

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

Issuance of this amendment is requested no later than October 4, 2013, with the amendment to be implemented within 90 days.

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

A001
NRR

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

Sincerely,

Dan Stoddard

Daniel G. Stoddard
Senior Vice President – Nuclear Operations

COMMONWEALTH OF VIRGINIA
COUNTY OF HENRICO

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Daniel G. Stoddard, who is Vice President – Nuclear Operations of Dominion Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 4th day of October, 2012.

My Commission Expires: 12-31-2016

Craig D Sly
Notary Public

CRAIG D SLY
Notary Public
Commonwealth of Virginia
Reg. # 7518653
My Commission Expires December 31, 2016

Attachments:

1. Description and Assessment of Proposed Changes
2. Documentation of PRA Technical Adequacy
3. Marked-up Technical Specifications Changes
4. Cross-References - NUREG-1431 to MPS3 TS Surveillance Frequencies Removed
5. Significant Hazards Consideration Determination
6. Marked-Up Technical Specifications Bases Changes (For Information Only)

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
Region I Regional Administrator
2100 Renaissance Blvd., Suite 100
King of Prussia, PA 19406-2713

J. S. Kim
NRC Project Manager
U.S. Nuclear Regulatory Commission, Mail Stop 08 C2A
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Millstone Power Station

Director, Radiation Division
Department of Energy and Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

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 only.

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).

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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 STI changes within the proposed licensee controlled program.

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

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

- 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.
   - Identify 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, which includes only the 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, M310A, and associated documentation have 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 M310A 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.

PRA Maintenance and Update

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

There are several procedures and GARDs (Guidance and Reference 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.

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 modifications and procedure changes are reviewed on an approximate quarterly or more frequent basis to determine if they impact the PRA and if a PRA modeling and/or documentation change is warranted. These reviews are documented, and if any PRA changes are warranted, they are added to the PRA Configuration Control (PRACC) database for PRA implementation tracking.

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

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

Date	Model Change (continued)
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	Large Early Release Frequency (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 90°F
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)
2012	Model update (M310A) a. updated with plant-specific data b. addressed several not-met supporting requirements c. enhanced documentation

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 LERF.

The findings and observations (F&Os) from the PRA peer review were prioritized into four categories (A through D) based upon importance to the completeness of the model. All F&Os identified during the 1999 WOG peer review 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].

Many of the supporting requirements (SRs) identified in the self-assessment as not meeting Capability Category II were incorporated into the MPS3 2008 PRA model (M308A) revision. 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.

MPS3 2012 Focused PRA Peer Review

The MPS3 PRA model was then updated (M310A) to reflect the current plant configuration and accumulation of additional plant operating history and component failure data. In the M310A model update, nearly all of the remaining not-met SRs were addressed by further upgrades to the model documentation as well as improvements to the model.

In June 2012, Science Applications International Corporation (SAIC) performed a focused PRA peer review of model upgrades [Ref. 10] incorporated since the 1999 WOG peer review. The purpose of the PRA peer review is to assess whether PRA upgrades, as defined by the ASME/ANS (American Nuclear Society) PRA standard, meet the intent of Category II SRs.

The scope of this review is defined in Table 1 [Ref. 9].

Table 1
Scope of MPS3 Focused Review

Technical Element	High Level Requirements (HLRs)	Supporting Requirements (SRs) Covered	Comments
IE	IE-C	IE-C8, IE-C9, IE-C10, IE-C11, IE-C14	Focused on support system initiating event fault tree models including loss of single alternating current and direct current buses and reactor plant ventilation, and the Interfacing Systems Loss of Coolant Accident (ISLOCA) analysis.
AS	AS-A	All	
	AS-B	All	
	AS-C	All	
SC	SC-B	All	Comprehensive review with attention to revised power operated relief valve success criteria for bleed & feed (BAF) for sequences where steam generator (SG) cooling is lost late (after demineralized water storage tank (DWST) depletion).
SY	SY-A	SY-A6, SY-A8, SY-A11, SY-A14, SY-A18	Focused on cooling dependency for the charging pump oil cooling system (CCE), ventilation dependencies, and methodology change from "black box" to component boundaries for the reactor protection system and engineered safeguards features actuation system.
	SY-B	SY-B5, SY-B6, SY-B7, SY-B9, SY-B10	
HR	HR-C	All	Focused on HRA updated to current plant procedures and timing, and rescreening of pre-initiator HEPs and their updated probabilities using THERP.
	HR-D	All	
	HR-G	All	
DA	DA-B	All	Focused on the changed method for common cause failures from Multiple Greek Letter to the Alpha factor method, and the update to include a systematic approach to grouped common cause failure treatment.
	DA-D	DA-D5, DA-D6	
LE	All	All	
IF	All	All	

The ASME/ANS PRA standard [Ref. 8] contains a total of 316 numbered SRs for internal events and internal flooding in nine technical elements. The focused scope of this review covered a total of 166 SRs. Three (3) of the SRs were determined to be not applicable to the MPS3 PRA model. Of the 163 SRs, 156 SRs, or 95.7%, were rated as SR Met, Capability Category II, or greater. None of the SRs were rated as only Category I, but seven (7) SRs were Not Met. A listing of the Not Met SRs is provided in Table 2 and evaluated in the gap analysis provided in Table 3.

Table 2
SRs Assessed as Not Met or Category I for the MPS3 PRA

Technical Element	Not Met SRs	Cat I SRs
Initiating Event (IE)	None	None
Accident Sequence Analysis (AS)	None	None
Success Criteria (SC)	None	None
Systems Analysis (SY)	SY-B6*	None
Human Reliability Analysis (HR)	None	None
Data Analysis (DA)	None	None
Internal Flooding (IF)	IFPP-B2	None
	IFSO-A4	
	IFSN-A8	
	IFEV-A5	
	IFEV-A7	
Large Early Release Frequency (LE)	LE-D2	None

* Documentation issue that was resolved following the peer review

Gaps to Meeting the ASME/ANS Standard

Table 3 provides a list of "gaps" in the technical adequacy of the MPS3 PRA model, M310A. Technical adequacy gaps were identified during the internal self-assessment and the focused peer review. For each gap, the table provides a gap description, ASME/ANS SR, Current Status/Comment and Importance to Application fields. Modeling gaps are classified as either high risk significance or low risk significance (based on their Fussell-Vesely (FV) importance value using a threshold value of 5E-3), or are classified as no impact to the application.

In accordance with NEI 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," open gaps that would impact the results of the SFCP PRA assessment will require sensitivity studies. These are identified in Table 3. These open gaps will be addressed in a PRA model periodic update in accordance with the PRA program procedural requirements.

Table 3
Status of Identified Gaps to Capability Category II of the ASME/ANS PRA Standard

Title	Capability Category II Requirements	ASME/ANS SR	Current Status / Comment	Importance to Application
Gap #1	Interview plant personnel (e.g., Operations, Maintenance, Safety Analysis) to determine if potential initiating events have been overlooked.	IE-A8*	There is currently no documentation of previous informal discussion between plant personnel and the PRA engineers on modeled initiating events in the MPS3 PRA. The PRA staff, which include previous system engineers, senior reactor operators and shift technical advisors, have worked closely over the years with plant personnel in support of risk-informed activities (e.g., Maintenance Rule (a)(4), Mitigating System Performance Indicators, Notice of Enforcement Discretion, and Significance Determination Process). A process has been developed and implemented to document discussions with plant personnel going forward.	**Not Risk Significant** Documentation issue only. Following implementation of the SFCP and in accordance with PRA procedures, a sensitivity study will not be required.
Gap #2	Perform plant walkdowns and interviews with knowledgeable plant personnel to confirm system analysis correctly reflects the as-built, as-operated plant.	SY-A4*	A process has been developed and implemented to document additional information on plant walkdowns and plant personnel interviews within the system notebooks. This is an on-going process. Also see Current Status/Comment for Gap #1.	**Not Risk Significant** Documentation issue only. Following implementation of the SFCP and in accordance with PRA procedures, a sensitivity study will not be required.

Table 3
Status of Identified Gaps to Capability Category II of the ASME/ANS PRA Standard

Title	Capability Category II Requirements	ASME/ANS SR	Current Status / Comment	Importance to Application
Gap #3	When needed, BASE the required time to complete actions for significant Human Failure Events (HFEs) on action time measurements in either walkthroughs or talk-throughs of the procedures or simulator observations.	HR-G5**	Existing Human Reliability Analysis documentation for talk-throughs with Operations is outdated (circa 2006). No new operator survey information is provided to support the basis for revised or new HFEs.	**High Risk Significant** Several potentially impacted HEPs are considered risk significant. In accordance with the SFCP, a bounding sensitivity study will be performed by increasing the HEP by a factor of 10 for all new and revised HEPs without talk-through with Operations.
Gap #4	For multiple human actions in the same accident sequence or cut set, identified in accordance with supporting requirement QU-C1, ASSESS the degree of dependence, and calculate a joint HEP that reflects the dependence. ACCOUNT for the influence of success or failure in preceding human actions and system performance on the human event under consideration including (a) the time required to complete all actions in relation to the time available to perform the actions (b) factors that could lead to dependence (e.g., common instrumentation, common procedures, increased stress) (c) availability of resources (e.g., personnel)	HR-G7**	There are several numerical inconsistencies in the dependency analysis spreadsheet supporting the HEP Dependency Analysis Notebook, HR.4.	**High Risk Significant:** This is considered high risk significance due to the importance of operators backing up automatic functions and some manual actions (e.g., hot leg injection). In accordance with the SFCP, an internal review of the dependency analysis spreadsheet will be conducted. Any HEPs identified to be incorrectly calculated will require a bounding sensitivity study by increasing the failure probability by a factor of 10.

Table 3
Status of Identified Gaps to Capability Category II of the ASME/ANS PRA Standard

Title	Capability Category II Requirements	ASME/ANS SR	Current Status / Comment	Importance to Application
Gap #5	DOCUMENT the process used to identify flood areas. For example, this documentation typically includes *(a)* flood areas used in the analysis and the reason for eliminating areas from further analysis *(b)* any walkdowns performed in support of the plant partitioning.	IFPP-B2	Most flood areas are based on the Fire Hazards Analysis, however, some areas are partitioned into smaller areas or split between multiple areas without a specific relationship to the Fire Hazards Analysis. Include additional documentation on flood propagation paths between the two operating units and the condensate polishing facility (CPF). For example, it is possible for water to propagate from the CPF to the turbine building. Potential water propagation is bounded by the already analyzed internal flooding events in the turbine building. Specifically, the amount of water generated during a circulating water pipe break is far greater than the amount of water possibly generated in the CPF.	**Low Risk Significance:** In accordance with the SFCP, a bounding sensitivity study will be performed by doubling the circulating water initiating event frequencies.

Table 3
Status of Identified Gaps to Capability Category II of the ASME/ANS PRA Standard

Title	Capability Category II Requirements	ASME/ANS SR	Current Status / Comment	Importance to Application
Gap #6	For each potential source of flooding, IDENTIFY the flooding mechanisms that would result in a release. INCLUDE: (a) failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc. (b) human-induced mechanisms that could lead to overfilling tanks, diversion of flow-through openings created to perform maintenance; inadvertent actuation of fire-suppression system (c) other events resulting in a release into the flood area.	IFSO-A4	Need to incorporate internal flooding frequencies associated with non-piping failures (e.g., expansion joints, bellows, overfill, and inadvertent sprinkler actuation). The majority of non-piping components (e.g., pumps, valves, tanks) are identified and included in the internal flooding analysis. The remaining non-piping failures (expansion joints, bellows and inadvertent actuation of fire protection systems) are bounded by already analyzed flow rates. Since the remaining non-piping failures make up a small percentage of the overall system piping failures, any changes in the internal flooding initiating event frequencies will not have a significant impact to the overall core damage frequency (CDF)/LERF or impact the SFCP.	**Low Risk Significance:** Internal flooding contributes only ~2% to the overall CDF. Any potential changes in flooding frequencies will only result in very small impact to the overall CDF. Following implementation of the SFCP and in accordance with PRA procedures, a bounding sensitivity study will be performed by doubling the internal flooding initiating event frequencies.
Gap #7	COMPARE results and EXPLAIN differences in the initiating event analysis with generic data sources to provide a reasonableness check of the results.	IE-C12*	Perform a reasonableness check of the expansion joint rupture frequencies modeled in the PRA.	**Low Risk Significance:** Internal flooding contributes only ~2% to the overall CDF. Any potential changes in expansion joint rupture frequencies will only result in very small impact to the overall CDF (See Gap #6). Following implementation of the SFCP and in accordance with PRA procedures, a bounding sensitivity study will be performed by doubling the expansion joint frequencies.

Table 3
Status of Identified Gaps to Capability Category II of the ASME/ANS PRA Standard

Title	Capability Category II Requirements	ASME/ANS SR	Current Status / Comment	Importance to Application
Gap #8	For each defined flood area and each flood source, IDENTIFY those automatic or operator responses that have the ability to terminate or contain the flood propagation.	IFSN-A3**	There is a mismatch between some of the internal flooding operator actions discussed in PRA Notebooks, *Flood Scenario Development* IF.2 and *Recovery Action Analysis for Internal Flooding Events* HR.10.	**Not Risk Significant** Documentation issue only. Following implementation of the SFCP and in accordance with PRA procedures, a sensitivity study will not be required.
Gap #9	IDENTIFY inter-area propagation through the normal flow path from one area to another via drain lines; and areas connected via backflow through drain lines involving failed check valves, pipe and cable penetrations (including cable trays), doors, stairwells, hatchways, and heating, ventilation and air conditioning ducts. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads.	IFSN-A8	Internal Flooding notebook, IF.2, does not consider the potential for barrier unavailability. For example, "...*Flood Compartment CSW-3. The room is equipped with a water tight door that remains intact, and thus propagation to other compartments is not postulated.*" In addition, no mention of floor drain check valves are included in the Internal Flooding notebooks.	**Low Risk Significance** Following implementation of the SFCP and in accordance with PRA procedures, a sensitivity study will be performed that will investigate the potential flood propagation via barrier unavailability or floor drain failure that could impact the surveillance frequency change.
Gap #10	DETERMINE the flood initiating event frequency for each flood scenario group by using the applicable requirements in 2-2.1.	IFEV-A5	In the current PRA model, M310A, the internal flooding initiating events are in "per calendar year". This is conservative since the Internal Flooding frequency has not been multiplied by capacity factor. Additionally, Internal Flooding only contributes ~2% to the overall CDF value.	**Not Risk Significant** Conservative, non-risk significant contributions to the CDF. Following implementation of the SFCP and in accordance with PRA procedures, a sensitivity study will not be required.

Table 3
Status of Identified Gaps to Capability Category II of the ASME/ANS PRA Standard

Title	Capability Category II Requirements	ASME/ANS SR	Current Status / Comment	Importance to Application
Gap #11	INCLUDE consideration of human-induced floods during maintenance through application of generic data.	IFEV-A7	Process for identifying human-induced flooding scenarios is not documented. However, the MPS3 PRA does include four human-induced flooding scenarios. Revise the flooding analysis to document the process used to identify human-induced flood scenarios.	**Not Risk Significant** Documentation issue only. Following implementation of the SFCP and in accordance with PRA procedures, a sensitivity study will not be required.
Gap #12	EVALUATE the impact of containment seals, penetrations, hatches, drywell heads (boiling water reactors), and vent pipe bellows and INCLUDE as potential containment challenges, as required. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	LE-D2	An analysis of potential penetration failures was performed as part of the 1983 MPS3 Plant Safety Study, which is likely outdated based on research conducted after 1983. The containment capacity analysis should consider degradation of seal performance at elevated temperatures based on newer research information. NUREG/CR-6906 [Ref. 11] reports that compression seals and gaskets, electrical penetration assemblies, and personnel airlocks were shown to fail when tested in excess of DBA pressures and temperatures by a factor of 2 to 5. This is consistent with the conclusions in the MPS3 Level 2 analysis.	**Not Risk Significant** Documentation issue only. Following implementation of the SFCP and in accordance with PRA procedures, a sensitivity study will not be required.

* Not Met SR identified during the self-assessment process which did not meet the requirement for a peer review (i.e., was not categorized as a model upgrade since the 1999 WOG peer review).

** These SRs were categorized as Met by the peer review team. However, an F&O was written for the SR based on non-systematic discrepancies that the PRA peer review team judged to require correction.

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 STI change.

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 fire risk assessment performed for the IPEEE must be qualitatively assessed for impact of the STI extension on fire risk. The IPEEE fire risk analysis quantified a 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 IPEEE. Therefore, the results of the seismic risk assessment performed for the IPEEE must be qualitatively assessed for impact of the STI extension on seismic event risk. The IPEEE 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 unavailability. 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 emergency diesel generator 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 Events

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 used.

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 SFCP. In performing the assessments for the other hazard groups, the qualitative or bounding approach will be utilized in most cases. Also, in addition to the standard set of sensitivity studies required per the NEI 04-10, Revision 1, methodology, open items for changes at the site and remaining gaps to specific requirements in the ASME/ANS PRA standard will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

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 2007

8. ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" and its 2009 addendum (ASME/ANS RA-Sa-2009).
9. Millstone Power Station Unit 3 Probabilistic Risk Assessment Model Notebook Part IV, Appendix B, "Quality Summary," May 2012.
10. Millstone Power Station Unit 3 Probabilistic Risk Assessment Model Notebook Part IV, Appendix A.3, "Reg Guide 1.200 Peer Review," August 2012.
11. NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories," July 2006.

ATTACHMENT 3

MARKED-UP TECHNICAL SPECIFICATIONS CHANGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

THIS IS A CONTROLLED COPY OF THE UNIT 3 TECHNICAL SPECIFICATIONS
CURRENT THROUGH CHANGE NO. 263
UPDATED BY LISA SCRUGGS

TABLE 1.1
FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.
SFCP	At the frequency specified in the Surveillance Frequency Control Program

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.

SURVEILLANCE REQUIREMENTS (Continued)

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

1) Reactor Coolant System boron concentration,
2) Control rod position,
3) Reactor Coolant System average temperature,
4) Fuel burnup based on gross thermal energy 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~~.

the frequency specified in the Surveillance Frequency Control Program

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

REACTIVITY CONTROL SYSTEMS

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,

* 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.

LIMITING CONDITION FOR OPERATION

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~~ the frequency specified in the Surveillance Frequency Control Program except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

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

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

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

a. T_{avg} greater than or equal to 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

REACTIVITY CONTROL SYSTEMS

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

* See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

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

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~~ the frequency specified in the Surveillance Frequency Control Program except during time intervals when the rod insertion limit monitor is inoperable, ~~then verify~~ the individual rod positions at least once per 4 hours.

* See Special Test Exceptions 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:

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

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

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.

e. With the maximum value of

the frequency specified in the Surveillance Frequency Control Program

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

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:

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

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

LIMITING CONDITION FOR OPERATION

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

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits 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:

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.~~

(8) (Not used) Deleted

(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.

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the 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 "SFCP"

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)								
c. 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.

INSTRUMENTATION

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.

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

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.

INSTRUMENTATION

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.

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

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

TABLE 4.3-6
REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	~~M~~	N.A.
2. Pressurizer Pressure	~~M~~	~~R~~
3. Pressurizer Level	~~M~~	~~R~~
4. Steam Generator Pressure	~~M~~	~~R~~
5. Steam Generator 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

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

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			

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

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.

INSTRUMENTATION

3/4.3.5 SHUTDOWN MARGIN MONITOR (continued)

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.

REACTOR COOLANT SYSTEM

HOT STANDBY

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

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

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.

REACTOR COOLANT SYSTEM

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

REACTOR COOLANT SYSTEM

HOT STANDBY

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

REACTOR COOLANT SYSTEM

RELIEF VALVES

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

REACTOR COOLANT SYSTEM

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 frequency specified in the Surveillance Frequency Control Program

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

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

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~~ the frequency specified in the Surveillance Frequency Control Program 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.

d. Performance of a Reactor Coolant System water inventory balance at ~~least once per 72 hours~~ the frequency specified in the Surveillance Frequency Control Program;

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~~ the frequency specified in the Surveillance Frequency Control Program, and;

f. Monitoring the Reactor Head Flange Leakoff System at ~~least once per 24 hours~~ the frequency specified in the Surveillance Frequency Control Program.

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.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS (Continued)

a. At ~~least once per 24 months~~ the frequency specified in the Surveillance Frequency Control Program,

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

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

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEM

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~~ 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

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

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

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A 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

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At ~~least once per 7 days~~ the frequency specified in the Surveillance Frequency Control Program by:

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

2) Verifying the boron concentration of the water.

b. At ~~least once per 24 hours~~ the frequency specified in the Surveillance Frequency Control Program 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~~ the frequency specified in the Surveillance Frequency Control Program by verifying that a minimum total of 974 cubic feet of TSP is contained in the TSP Storage Baskets.

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:

a. At ~~least once per 31 days~~ the frequency specified in the Surveillance Frequency Control Program 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

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(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.

CONTAINMENT AIR LOCKS

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

CONTAINMENT SYSTEMS

RECIRCULATION SPRAY SYSTEM

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 the frequency specified in the Surveillance Frequency Control Program ~~least once per 24 months~~ 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~~ the frequency specified in the Surveillance Frequency Control Program.

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.

CONTAINMENT SYSTEMS

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%:

d. At ~~least once per 24 months~~ the frequency specified in the Surveillance Frequency Control Program 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.

CONTAINMENT SYSTEMS

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

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

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

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At ~~least once per 31 days~~ by: the frequency specified in the Surveillance Frequency Control Program

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

Auxiliary feedwater pumps may be considered OPERABLE during alignment and operation for steam generator level control, if they are capable of being manually realigned to the auxiliary feedwater mode of operation.

- -

Verifying each auxiliary feedwater manual, power operated, and automatic valve in each water flow path and in each required steam supply flow path to the steam turbine driven auxiliary feedwater pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At ~~least once per 92 days on a STAGGERED TEST BASIS~~, tested pursuant to Specification 4.0.5, by:

1) Verifying that on recirculation flow each motor-driven pump develops a total head of greater than or equal to 3385 feet;

2) Verifying that on recirculation flow the steam turbine-driven pump develops a total head of greater than or equal to 3780 feet when the secondary steam supply pressure is greater than 800 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

SURVEILLANCE REQUIREMENTS (Continued)

c. At ~~least once per 24 months~~ by verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal. For the steam turbine-driven auxiliary feedwater pump, the provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 2 by verifying flow to each steam generator.

the frequency specified in the Surveillance Frequency Control Program

PLANT SYSTEMS

DEMINERALIZED WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The demineralized water storage tank (DWST) shall be OPERABLE with a water volume of at least 334,000 gallons.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the DWST inoperable, within 4 hours either:

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

b. Demonstrate the OPERABILITY of the condensate storage tank (CST) as a backup supply to the auxiliary feedwater pumps and restore the DWST to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The DWST shall be demonstrated OPERABLE at ~~least once per 12 hours~~ by verifying the water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The CST shall be demonstrated OPERABLE at ~~least once per 12 hours~~ by verifying that the combined volume of both the DWST and CST is at least 384,000 gallons of water whenever the CST and DWST are the supply source for the auxiliary feedwater pumps.

the frequency specified in the Surveillance Frequency Control Program

TABLE 4.7-1
SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

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

At the frequency specified in the Surveillance Frequency Control Program

PLANT SYSTEMS

STEAM GENERATOR ATMOSPHERIC RELIEF BYPASS LINES

LIMITING CONDITION FOR OPERATION

3.7.1.6 Each steam generator atmospheric relief bypass valve (SGARBV) line shall be OPERABLE, with the associated main steam atmospheric relief isolation (block) valve in the open position.

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

ACTIONS

a. With one required SGARBV line inoperable, restore required SGARBV line to OPERABLE status within 7 days or be in at least MODE 3 within the next 6 hours and be in MODE 4 without reliance upon steam generator for heat removal within the next 18 hours. LCO 3.0.4 is not applicable.

b. With two or more required SGARBV lines inoperable, restore all but one required SGARBV line to OPERABLE status within 24 hours or be in at least MODE 3 within the next 6 hours and be in MODE 4 without reliance upon steam generator for heat removal within the next 18 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6.1 Verify one complete cycle of each SGARBV ~~every 18 months~~.

4.7.1.6.2 Verify one complete cycle of each main steam atmospheric relief isolation (block) valve ~~every 18 months~~.

at the frequency specified in the Surveillance Frequency Control Program

PLANT SYSTEMS

3/4.7.3 REACTOR PLANT COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

b. At ~~least once per 24 months~~ by verifying that:

1) Each automatic valve actuates to its correct position on its associated Engineered Safety Feature actuation signal, and

2) Each Component Cooling Water System pump starts automatically on an SIS test signal.

the frequency specified in the Surveillance Frequency Control Program

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

b. At ~~least once per 24 months~~ by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation signal, and

2) Each Service Water System pump starts automatically on an SIS test signal.

the frequency specified in the Surveillance Frequency Control Program

PLANT SYSTEMS

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink (UHS) shall be OPERABLE with an average water temperature of less than or equal to 75°F.

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

ACTION:

If the UHS temperature is above 75°F, monitor the UHS temperature once per hour for 12 hours. If the UHS temperature does not drop below 75°F during this period, place the plant in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. During this period, if the UHS temperature increases above 77°F, place the plant in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The UHS shall be determined OPERABLE:

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

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

the frequency specified in the Surveillance Frequency Control Program

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

e. With both Control Room Emergency Air Filtration Systems inoperable, or with the OPERABLE Control Room Emergency Air Filtration System required to be in the emergency mode by ACTION d. not capable of being powered by an OPERABLE emergency power source, or with one or more Control Room Emergency Air Filtration System Trains inoperable due to an inoperable CRE boundary, immediately suspend the movement of recently irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.7 Each Control Room Emergency Air Filtration System shall be demonstrated OPERABLE:

the frequency specified in the Surveillance Frequency Control Program

a. At ~~least once per 12 hours~~ by verifying that the control room air temperature is less than or equal to 95°F;

b. At ~~least once per 31 days on a STAGGERED TEST BASIS~~ by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying a system flow rate of 1,120 cfm ± 20% and that the system operates for at least 10 continuous hours with the heaters operating;

c. 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 Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revisions 2, March 1978,* and the system flow rate is 1,120 cfm ± 20%;

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), a relative humidity of 70%, and a face velocity of 54 ft/min; and

3) Verifying a system flow rate of 1,120 cfm ± 20% during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* 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 a face velocity of 54 ft/min.

e. At ~~least once per 24 months~~ the frequency specified in the Surveillance Frequency Control Program by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.75 inches Water Gauge while operating the system at a flow rate of 1,120 cfm ± 20%;

2) Deleted

3) Verifying that the heaters dissipate 9.4 ±1 kW when tested in accordance with ANSI N510-1980.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 1120 cfm ± 20%; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 1120 cfm ± 20%.

h. By performance of CRE unfiltered air inleakage testing in accordance with the CRE Habitability Program at a frequency in accordance with the CRE Habitability Program.

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

PLANT SYSTEMS

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9 Two independent Auxiliary Building Filter Systems shall be OPERABLE.

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

ACTION:

With one Auxiliary Building Filter 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. In addition, comply with the ACTION requirements of Specification 3.6.6.1.

SURVEILLANCE REQUIREMENTS

the frequency specified in the Surveillance Frequency Control Program

4.7.9 Each Auxiliary Building Filter 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 30,000 cfm ±10% 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 cleanup 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 30,000 cfm ±10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* 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), a relative humidity of 70%, and a face velocity of 52 ft/min; and

3) Verifying a system flow rate of 30,000 cfm ±10% 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), a relative humidity of 70%, and a face velocity of 52 ft/min;

d. At the frequency specified in the Surveillance Frequency Control Program ~~least once per 24 months~~ by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.8 inches Water Gauge while operating the system at a flow rate of 30,000 cfm ±10%,

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

3) Verifying that the heaters dissipate 180 ±18 kW when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 30,000 cfm ±10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 30,000 cfm ±10%.

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

LIMITING CONDITION FOR OPERATION

ACTION (continued)

Inoperable Equipment	Required ACTION
e. Two diesel generators	e.2 Restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. AND e.3 Following restoration of one diesel generator, restore remaining inoperable diesel generator to OPERABLE status following the time requirements of ACTION Statement b. above based on the initial loss of the remaining inoperable diesel generator.

the frequency specified in the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at ~~least once per 7 days~~ by verifying correct breaker alignments, indicated power availability, and

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

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:*

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

1) Verifying the fuel level in the day tank,

2) Verifying the fuel level in the fuel storage tank,

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank,

4) Verifying the lubricating oil inventory in storage,

5) Verifying the diesel starts from standby conditions and achieves generator voltage and frequency at 4160 ± 420 volts and 60 ± 0.8 Hz. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

* All planned starts for the purpose of these surveillances may be preceded by an engine prelube period.

the frequency specified in the Surveillance Frequency Control Program

b) Simulated loss-of-offsite power by itself, or

c) Simulated loss-of-offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

6) Verifying the generator is synchronized and gradually loaded in accordance with the manufacturer's recommendations between 4800-5000 kW* and operates with a load between 4800-5000 kW* for at least 60 minutes, and

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

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

1) Verifying that the diesel generator starts from standby conditionS and attains generator voltage and frequency of 4160 ± 420 volts and 60 ± 0.8 Hz within 11 seconds after the start signal.

2) Verifying the generator is synchronized to the associated emergency bus, loaded between 4800-5000 kW* in accordance with the manufacturer's recommendations, and operate with a load between 4800-5000 kW* for at least 60 minutes.

The diesel generator shall be started for this test using one of the signals in Surveillance Requirement 4.8.1.1.2.a.5. This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.5, may also serve to concurrently meet those requirements as well.

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

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

e. By sampling new fuel oil in accordance with ASTM-D4057 prior to addition to storage tanks and:

1) By verifying in accordance with the tests specified in ASTM-D975-81 prior to addition to the storage tanks that the sample has:

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

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* The operating band is meant as guidance to avoid routine overloading of the diesel. Momentary transients outside the load range shall not invalidate the test.

SURVEILLANCE REQUIREMENTS (Continued)

the frequency specified in the Surveillance Frequency Control Program

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

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

d) Water and sediment less than 0.05 percent by volume when tested in accordance with ASTM-D1796-83.

2. By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that: (1) the cetane index shall be determined in accordance with ASTM-D976 (this test is an appropriate approximation for cetane number as stated in ASTM-D975-81 [Note E]), and (2) the analysis for sulfur may be performed in accordance with ASTM-D1552-79, ASTM-D2622-82 or ASTM-D4294-83.

f. At ~~least once every 31 days~~ by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A;

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

1) DELETED

2) Verifying the generator capability to reject a load of greater than or equal to 595 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 3 Hz;

3) Verifying the generator capability to reject a load of 4986 kW without tripping. The generator voltage shall not exceed 5000 volts during and 4784 volts following the load rejection;

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b) Verifying the diesel starts from standby conditions on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 0.8 Hz during this test.

8) Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 5335 kW;

9) Verifying the diesel generator's capability to:

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

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

c) Be restored to its standby status.

10) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

11) DELETED

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval; and

13) DELETED

h. At ~~least once per 10 years~~ by starting both diesel generators simultaneously from standby conditions, during shutdown, and verifying that both diesel generators achieve generator voltage and frequency at 4160 ± 420 volts and 60 ± 0.8 Hz in less than or equal to 11 seconds; and

i. At ~~least once per 10 years~~ by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution.

the frequency specified in the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (Continued)

j. At ~~least once per 18 months~~ by verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded between 5400-5500kW* and during the remaining 22 hours of this test, the diesel generator shall be loaded between 4800-5000kW*. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 0.8 Hz within 11 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test.** Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2.a.5) excluding the requirement to start the diesel from standby conditions.***

k. At ~~least once per 18 months~~ by verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines.

l. At ~~least once per 18 months~~ by verifying that the following diesel generator lockout features prevent diesel generator starting:

1) Engine overspeed,

2) Lube oil pressure low (2 of 3 logic),

3) Generator differential, and

4) Emergency stop.

the frequency specified in the Surveillance Frequency Control Program

* The operating band is meant as guidance to avoid routine overloading of the diesel. Momentary transients outside the load range shall not invalidate the test.

** Diesel generator loadings may include gradual loading as recommended by the manufacturer.

*** If Surveillance Requirement 4.8.1.1.2.a.5) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated between 4800-5000 kW for 2 hours or until operating temperature has stabilized.

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

a. 125-volt Battery Bank 301A-1, and an associated full capacity charger,

b. 125-volt Battery Bank 301A-2, and an associated full capacity charger,

c. 125-volt Battery Bank 301B-1 and an associated full capacity charger, and

d. 125-volt Battery Bank 301B-2 and an associated full capacity charger.

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

ACTION:

a. With either Battery Bank 301A-1 or 301B-1, and/or one of the required full capacity chargers inoperable, restore the inoperable battery bank and/or full capacity charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With either Battery Bank 301A-2 or 301B-2 inoperable, and/or one of the required full capacity chargers inoperable, restore the inoperable battery bank and/or full capacity charger 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 REQUIREMENTS

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

a. At ~~least once per 7 days~~ the frequency specified in the Surveillance Frequency Control Program by verifying that:

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohm, and

3) The average electrolyte temperature of six connected cells is above 60°F.

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

the frequency specified in the Surveillance Frequency Control Program

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

2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm, and

4) Each battery charger will supply at least the amperage indicated in Table 4.8-2b at greater than or equal to 132 volts for at least 24 hours.

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

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e. At ~~least once per 60 months~~, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. ~~Once per 60-month interval~~ this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION

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

ACTION:

a. With one of the required trains of A.C. emergency busses not OPERABLE, restore the inoperable train to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

the frequency specified in the Surveillance Frequency Control Program

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION (Continued)

4) Two 125 volt DC Busses consisting of:

a) Bus #301B-1 energized from Battery Bank #301B-1, and

b) Bus #301B-2 energized from Battery Bank #301B-2.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity additions that could result in loss of required SDM or boron concentration, movement of recently irradiated fuel assemblies, crane operation with loads over the fuel storage pool, or operations with a potential for draining the reactor vessel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

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

the frequency specified in the Surveillance Frequency Control Program

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

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

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

b. A boron concentration of greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

Additionally, the CVCS valves of Specification 4.1.1.2.2 shall be closed and secured in position.

APPLICABILITY: MODE 6.*

ACTION:

a. With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and positive reactivity additions and 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 its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to the limit specified in the COLR, whichever is the more restrictive.

b. With any of the CVCS valves of Specification 4.1.1.2.2 not closed** and secured in position, immediately close and secure the valves.

SURVEILLANCE REQUIREMENTS

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

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.1.2 The boron concentration of the Reactor Coolant System and the refueling cavity shall be determined by chemical analysis at ~~least once per 72 hours~~.

4.9.1.1.3 The CVCS valves of Specification 4.1.1.2.2 ~~shall be~~ verified closed and locked at ~~least once per 31 days~~.

the frequency specified in the Surveillance Frequency Control Program

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

** Except those opened under administrative control.

REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.2 The soluble boron concentration of the Spent Fuel Pool shall be greater than or equal to 800 ppm.

APPLICABILITY:

Whenever fuel assemblies are in the spent fuel pool.

ACTION:

a. With the boron concentration less than 800 ppm, initiate action to bring the boron concentration in the fuel pool to at least 800 ppm within 72 hours, and

b. With the boron concentration less than 800 ppm, suspend the movement of all fuel assemblies within the spent fuel pool and loads over the spent fuel racks.

SURVEILLANCE REQUIREMENTS

4.9.1.2 Verify that the boron concentration in the fuel pool is greater than or equal to 800 ppm ~~every 7 days~~.

at the frequency specified in the Surveillance Frequency Control Program

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 Two Source Range Neutron Flux Monitors shall be OPERABLE with continuous visual indication in the control room, and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable immediately suspend all operations involving CORE ALTERATIONS and operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.1.

b. With both of the above required monitors inoperable determine the boron concentration of the Reactor Coolant System within 4 hours and at least once per 12 hours thereafter.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

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

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

the frequency specified in the Surveillance Frequency Control Program

* Neutron detectors are excluded from CHANNEL CALIBRATION.

REFUELING OPERATIONS

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

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

a. The equipment access hatch shall be either:

1. closed and held in place by a minimum of four bolts, or

2. open under administrative control * and capable of being closed and held in place by a minimum of four bolts,

b. A personnel access hatch shall be either:

1. closed by one personnel access hatch door, or

2. capable of being closed by an OPERABLE personnel access hatch door, under administrative control,* and

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

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed under administrative control.*

APPLICABILITY: During movement of fuel within the containment building.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4.a Verify each required containment penetrations is in the required status at ~~least once per 7 days~~.

4.9.4.b DELETED the frequency specified in the Surveillance Frequency Control Program

* Administrative controls shall ensure that appropriate personnel are aware that the equipment access hatch penetration, personnel access hatch doors and/or other containment penetrations are open, and that a specific individual(s) is designated and available to close the equipment access hatch penetration, a personnel access hatch door and/or other containment penetrations within 30 minutes if a fuel handling accident occurs. Any obstructions (e.g. cables and hoses) that could prevent closure of the equipment access hatch penetration, a personnel access hatch door and/or other containment penetrations must be capable of being quickly removed.

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With no RHR loop OPERABLE or in operation, suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.1 and suspend loading irradiated fuel assemblies in the core and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

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

the frequency specified in the Surveillance Frequency Control Program

* The RHR loop may be removed from operation for up to 1 hour per 8-hour period, provided no operations are permitted that could cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.1.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, 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 the boron concentration of LCO 3.9.1.1 and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

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

the frequency specified in the Surveillance Frequency Control Program

* The RHR loop may be removed from operation for up to 1 hour per 8-hour period, provided no operations are permitted that could cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.1.

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

the frequency specified in the Surveillance Frequency Control Program

REFUELING OPERATIONS

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at ~~least once per 7 days~~ the frequency specified in the Surveillance Frequency Control Program when irradiated fuel assemblies are in the fuel storage pool.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

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

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

the frequency specified in the Surveillance Frequency Control Program

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2.1 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

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

b. The limits of Specifications 3.2.2.1 and 3.2.3.1 are maintained and determined at the frequencies specified in Specification 4.10.2.1.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2.1 or 3.2.3.1 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2.1 and 3.2.3.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.10.2.1.2 The Surveillance Requirements of the below listed specifications shall be performed at ~~least once per 12 hours~~ during PHYSICS TESTS:

a. Specifications 4.2.2.1.2 and 4.2.2.1.3, and

b. Specification 4.2.3.1.2

the frequency specified in the Surveillance Frequency Control Program

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: MODE 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

3 4.10.3. The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at ~~least once per 30 minutes~~ during PHYSICS TESTS.

the frequency specified in the Surveillance Frequency Control Program

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at ~~least once per hour~~ the frequency specified in the Surveillance Frequency Control Program during STARTUP and PHYSICS TESTS.

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

PROCEDURES AND PROGRAMS (Continued)

RG 1.183), which were considered in completing the vulnerability assessments, are documented in the UFSAR/current licensing basis. Compliance with these RGs is consistent with the current licensing basis as described in the UFSAR and other licensing basis documents.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVs, operating at the flow rate required by the Surveillance Requirements, at a Frequency of 48 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

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

Insert 1

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

6.8.5 Written procedures shall be established, implemented and maintained covering Section I.E, Radiological Environmental Monitoring, of the REMODCM.

6.8.6 All procedures and procedure changes required for the Radiological Environmental Monitoring Program (REMP) of Specification 6.8.5 above shall be reviewed by an individual (other than the author) from the organization responsible for the REMP and approved by appropriate supervision.

Temporary changes may be made provided the intent of the original procedure is not altered and the change is documented and reviewed by an individual (other than the author) from the organization responsible for the REMP, within 14 days of implementation.

INSERTS FOR TECHNICAL SPECIFICATIONS MARKUPS

INSERT 1 *(for TS 6.8.4)*

i. Surveillance Frequency Control Program

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

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

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

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

ATTACHMENT 4

CROSS-REFERENCES - NUREG-1431 TO MPS3 TS SURVEILLANCE FREQUENCIES REMOVED

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

CROSS-REFERENCE NUREG-1431 TS SURVEILLANCE REQUIREMENT FREQUENCIES TO MILLSTONE UNIT 3 TS SURVEILLANCE REQUIREMENT FREQUENCIES REMOVED

Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
Shutdown margin		
Verify SDM in Modes 2 w/keff < 1 ,3 4, and 5	SR 3.1.1.1	---
Verify SDM in Modes 1 and 2	---	4.1.1.1.1
Verify SDM in Modes 3, 4, and 5 Loops Filled	---	4.1.1.1.2.1 .b
Verify Valve Position 3-CHS v305	---	4.1.1.1.2.2
Verify SDM in Cold Shutdown Loops Not Filled	---	4.1.1.2.1.b
Verify Valve Positions	---	4.1.1.2.2
Core Reactivity		
Verify Reactivity $\pm$1%	SR 3.1.2.1	4.1.1.1.2
Rod Group Alignment		
Verify Rod Position within Alignment	SR 3.1.4.1	4.1.3.1.1
Verify Rod Movement	SR 3.1.4.2	4.1.3.1.2
Verify Rod Drop Times	---	4.1.3.4.c
Shutdown Bank Insertion Limits		
Verify Insertion Limits	SR 3.1.5.1	4.1.3.5.b
Control Bank Insertion limit		
Verify Limits within COLR	SR 3.1.6.2	4.1.3.6
Verify Control Bank Rod Sequence and Overlap	SR 3.1.6.3	---
Position Indication System		
Verify Digital Rod Position Operable DRPI vs. Demand Position Indication System	---	4.1.3.2.1
Verify Digital Rod Position Operable DRPI vs. Demand Position Indication System Agree When Exercised	---	4.1.3.2.2
Physics Test Exceptions		
Verify RCS Loop Temperature	SR 3.1.8.2	4.10.3.3
Verify Thermal Power $\leq$5%	SR 3.1.8.3	4.10.3.1
Verify Thermal Power < 85%	---	4.10.2.1.1
Verify SDM	SR 3.1.8.4	4.10.1.1
Perform Specs 4.2.2.1.2, 4.2.2.1.3, and 4.2.3.1.2	---	4.10.2.1.2
Determine Thermal Power < P-7	---	4.10.4.1
$F_Q(Z)$ Limits - RAOC		
Verify $F_Q(Z)$ limits - measured	SR 3.2.1.1	4.2.2.1.2.d(2)
Verify $F_Q(Z)$ limits Base Load Operations - Measured	---	4.2.2.1.4.d(2)
Verify $F_Q^W(Z)$ limits	SR 3.2.1.2	---

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Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
$F^N_{\Delta H}$ Limits		
Verify $F^N_{\Delta H}(Z)$ limits	SR 3.2.2.1	4.2.3.1.2.b
AFD Limits - RAOC		
Verify AFD Within Limit	SR 3.2.3.1	4.2.1.1.1.a
Base Loaded Operations - Determine by Measurement the AFD for Each Operable Excore	---	4.2.1.1.3
Base Loaded Operations – Updated Target AFD	---	4.2.1.1.4
QPTR		
Verify QPTR by calculation	SR 3.2.4.1	4.2.4.1.a
Verify QPTR w/ incore detectors	SR 3.2.4.2	4.2.4.2
RPS Instrumentation		
Perform Channel Check	SR 3.3.1.1	Table 4.3-1 Channel Check Column
Perform Calorimetric – actual power adjust if > 2%	SR 3.3.1.2	Table 4.3-1 Functional Unit (FU) 2
Compare and Adjust NIS to Incore ≥ 3%	SR 3.3.1.3	Table 4.3-1, FU 2
Perform TADOT Rx Trip Breakers	SR 3.3.1.4	Table 4.3-1 FUs, 18 & 21
Perform Actuation Logic Test	SR 3.3.1.5	Table 4.3-1 FU, Unit 19
Calibrate NIS to Incore	SR 3.3.1.6	Table 4.3-1, FU 2
Perform COT - 184 days	SR 3.3.1.7	Table 4.3-1 Analog Channel Operational Test Column
Perform COT (MPS3 – Quarterly Frequency)	SR 3.3.1.8	Table 4.3-1 FU 6
Perform TADOT	SR 3.3.1.9	Table 4.3-1 TADOT column
Perform Channel Calibration w/time constants	SR 3.3.1.10	---
Perform Channel Calibration w/o neutron detectors	SR 3.3.1.11	Table 4.3-1, FUs 2, 3, 5, & 6
Perform Channel Calibration w/ RTDs	SR 3.3.1.12	---
Perform COT – 18 months	SR 3.3.1.13	Table 4.3-1, FU 17
Perform TADOT – 18 months	SR 3.3.1.14	Table 4.3-1, FUs 1 & 21
Verify Response Time	SR 3.3.1.16	4.3.1.2
ESFAS Instrumentation		
Perform Channel Check	SR 3.3.2.1	Table 4.3-2, Channel Check Column
Perform Actuation Logic Test – 92 days	SR 3.3.2.2	Table 4.3-2, FU 10
Perform Actuation Logic Test – 31 days	SR 3.3.2.3	Table 4.3-2, FUs 1.b,2.b,3.a.2,3.b.2,4.b,5.a&b,6.b,7.c
Perform Master Relay Test	SR 3.3.2.4	Table 4.3-2 Master Relay Test Column
Perform COT – 184 days (MPS3 – Quarterly Frequency)	SR 3.3.2.5	Table 4.3-2 Analog Channel Operational Test Column

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Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
Perform Slave Relay Test - 92 days	SR 3.3.2.6	Table 4.3-2 Slave Relay Test Column
Perform TADOT – 92 days	SR 3.3.2.7	---
Perform TADOT – 18 months	SR 3.3.2.8	Table 4.3-2, FUs 1.a, 2.a, 3.a.1, 3.b.1, 4.d.1, 4.d.2, 5.c, 6.a, 7.a, 7.b, and 9.c
Perform Channel Calibration	SR 3.3.2.9	Table 4.3-2 Channel Calibration Column
Verify Time Response	SR 3.3.2.10	4.3.2.2
Radiation Monitoring Instrumentation MPS3		
Perform Check, Calibrate, and Analog COT	---	4.3.3.1
PAM Instrumentation		
PAM Channel Check	SR 3.3.3.1	4.3.3.6.1
PAM Channel Calibration	SR 3.3.3.2	4.3.3.6.1
Remote Shutdown System		
Perform Channel Check	SR 3.3.4.1	4.3.3.5.1
Verify Control and Transfer Switch Function	SR 3.3.4.2	4.3.3.5.2
Perform Channel Calibration	SR 3.3.4.3	4.3.3.5.1
Perform TADOT of Reactor Trip Breaker	SR 3.3.4.4	---
Shutdown Margin Monitor - MPS3		
Perform Analog COT	---	4.3.5.a
Verify Monitor Count Rate	---	4.3.5.b
LOP EDG Start Instrumentation		
Perform Channel Check	SR 3.3.5.1	---
Perform TADOT	SR 3.3.5.2	Table 4.3-2 , FU 8
Perform Channel Calibration	SR 3.3.5.3	Table 4.3-2, FU 8
Perform Response Time Testing	-----	4.3.2.2
Containment Purge and Vent Isolation		
Perform Channel Check	SR 3.3.6.1	---
Perform Actuation Logic Test – 31 days	SR 3.3.6.2	---
Perform Master Relay Test – 3 days	SR 3.3.6.3	---
Perform Actuation – 92 days	SR 3.3.6.4	---
Perform Master Relay Test -92 days	SR 3.3.6.5	---
Perform COT	SR 3.3.6.6	---
Perform Slave Relay Test	SR 3.3.6.7	---
Perform TADOT	SR 3.3.6.8	4.6.3.2.c
Perform Channel Calibration	SR 3.3.6.9	---
CREFS (*Control Building Isolation*)		
Perform Channel Check	SR 3.3.7.1	Table 4.3-2, FUs 7.d & e

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Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
Perform COT	SR 3.3.7.2	Table 4.3-2, FUs 7.d & e
Perform Actuation Logic Test – 31 days	SR 3.3.7.3	Table 4.3-2, FU 7.c
Perform Master Relay Test – 31 days	SR 3.3.7.4	Table 4.3-2, FU 7.c
Perform Actuation Logic Test – 92 days	SR 3.3.7.5	---
Perform Master Relay Test – 92 days	SR 3.3.7.6	---
Perform Slave Relay Test	SR 3.3.7.7	Table 4.3-2 ,FU 7.c
Perform TADOT	SR 3.3.7.8	Table 4.3-2 , FUs 7.a & b
Perform Channel Calibration	SR 3.3.7.9	Table 4.3-2, FUs 7.a & b
FBACS Actuation Instrumentation		
Perform Channel Check	SR 3.3.8.1	Note 1
Perform COT	SR 3.3.8.2	Note 1
Perform Actuation Logic Test	SR 3.3.8.3	Note 1
Perform TADOT	SR 3.3.8.4	Note 1
Perform Channel Calibration	SR 3.3.8.5	Note 1
BDPS *(Shutdown Monitor)*		
Perform Channel Check	SR 3.3.9.1	---
Perform COT	SR 3.3.9.2	---
Perform Channel Calibration	SR 3.3.9.3	---
RCS Press Temp & Flow Limits		
Verify Pressurizer Pressure	SR 3.4.1.1	4.2.5
Verify RCS Average Temperature	SR 3.4.1.2	4.2.5
Verify RCS Total Flow	SR 3.4.1.3	4.2.3.1.3.b
Verify RCS Total Flow w/ Heat Balance	SR 3.4.1.4	---
Calibrate RCS Total Flow Indicators	---	4.2.3.1.4
RCS Minimum Temp for Criticality		
Verify RSC Average Temperature in Each Loop	SR 3.4.2.1	---
RCS Temperature, Pressure,		
Verify Limits	SR 3.4.3.1	4.2.5
Loop Operation - Modes 1 and 2		
Verify Each Loop Operating	SR 3.4.4.1	4.4.1.1
Loop Operation - Mode 3		
Verify Required Loops Operating	SR 3.4.5.1	4.4.1.2.3
Verify Steam Generator Water Level ≥ 17%	SR 3.4.5.2	4.4.1.2.2
Verify Breaker Alignment and Power Available	SR 3.4.5.3	4.4.1.2.1
Loop Operation - Mode 4		
Verify Loop Operation – RHR or RCS	SR 3.4.6.1	4.4.1.3.3
Verify Steam Generator Water Level ≥ 17%	SR 3.4.6.2	4.4.1.3.2
Verify Breaker Alignment and Power Available	SR 3.4.6.3	4.4.1.3.1

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Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
Loop Operation - Mode 5 –Loops Filled		
Verify RHR Loop Operating	SR 3.4.7.1	4.4.1.4.1.2
Verify Steam Generator water Level ≥ 17%	SR 3.4.7.2	4.4.1.4.1.1
Verify Breaker Alignment and Power Available RHR Pumps	SR 3.4.7.3	4.4.1.4.1.3
Verify Loop Operation - Mode 5 –Loops - Not Filled		
Verify RHR Loop Operating	SR 3.4.8.1	4.4.1.4.2.2
Verify Breaker Alignment and Power Available RHR Pumps	SR 3.4.8.2	4.4.1.4.2.1
Pressurizer (Modes 1 and 2/Mode 3)		
Verify Water Level	SR 3.4.9.1	4.4.3.1.1/4.4.3.2.1
Verify Heater Capacity of Required Groups	SR 3.4.9.2	4.4.3.1.2/4.4.3.2.2
Verify Heater banks can be Powered from Emergency Power Supply	SR 3.4.9.4	---
Pressurizer PORVS		
Cycle each Block Valve	SR 3.4.11.1	4.4.4.2
Cycle each PORV	SR 3.4.11.2	4.4.4.1.b
Cycle each SOV Valve and Check Valve on the Air Accumulators in PORV Control Systems	SR 3.4.11.3	---
Verify PORVs and Block Valves can be Powered from Emergency Power Sources	SR 3.4.11.4	---
Perform Channel Calibration	---	4.4.4.1.a
Perform ACOT on PORV High Pressurize Pressure	---	4.4.4.1.c
Verify High Pressure Auto Open is Enabled	---	4.4.4.1.d
LTOP Systems		
Verify only one HPI pump is capable of injecting into the RCS.	SR 3.4.12.1	4.4.9.3.4
Verify a maximum of one charging pump is capable of injecting into the RCS.	SR 3.4.12.2	4.4.9.3.5
Verify each accumulator is isolated.	SR 3.4.12.3	---
Verify each RHR Suction Valve is open for each Relief Valve	SR 3.4.12.4	4.4.9.3.2.a
Verify required RCS vent [2.07] square inches open	SR 3.4.12.5	4.4.9.3.3
Verify PORV block valve is open for each required PORV.	SR 3.4.12.6	4.4.9.3.1.c
Verify PORV COPPS Armed	---	4.4.9.3.1.c
Verify RHR Suction Isolation Valve is Locked Open with Operator Power Removed for Required RHR Suction Relief Valve.	SR 3.4.12.7	---
Perform COT on each Required PORV	SR 3.4.12.8	4.4.9.3.1.a
Perform Channel Calibration on each Required PORV Channel	SR 3.4.12.9	4.4.9.3.1.b

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Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
Operational Leakage		
Verify RCS Operational Leakage	SR 3.4.13.1	4.4.6.2.1.d
Verify SG Leakage ≤150 gpd	SR 3.4.13.2	4.4.6.2.1.e
Monitor Reactor Head Flange Leakoff System	---	4.4.6.2.1.f
Measure Leakage to RCP Seals	---	4.4.6.2.1.c
RCS PIVs		
Verify leakage from each is ≤ 0.5 gpm	SR 3.4.14.1	4.4.6.2.2.a
Verify RHR Autoclosure Interlock Prevents Opening	SR 3.4.14.2	4.5.2.d.1
Verify RHR Autoclosure Interlock Auto Close	SR 3.4.14.3	---
RCS Leakage Detection Instrumentation		
Perform Channel Check – Particulate Rad Monitor	SR 3.4.15.1	4.4.6.1.a
Perform COT – Particulate Rad Monitor	SR 3.4.15.2	4.4.6.1.a
Perform Channel Calibration Sump Monitor	SR 3.4.15.3	4.4.6.1.b
Perform Channel Calibration containment atmosphere radioactivity monitor.	SR 3.4.15.4	4.4.6.1.a
Perform Channel Calibration containment air cooler.	SR 3.4.15.5	---
RCS Specific Activity		
Verify RCS gross specific activity	SR 3.4.16.1	---
Verify reactor coolant Dose Equivalent 1-131	SR 3.4.16.2	4.4.8.2
Determine E Bar	SR 3.4.16.3	---
Verify Xe-133	---	4.4.8.1
RCS Loop Isolation Valves		
Verify Open and Power Remove from Isolation Valves	SR 3.4.17.1	4.4.1.5
RCS Loops Test Exceptions		
Verify power < P-7	SR 3.4.19.1	4.10.4.1
Accumulators		
Verify Accumulator isolation valve open	SR 3.5.1.1	4.5.1.a.2)
Verify borated Water Volume	SR 3.5.1.2	4.5.1.a.1)
Verify N^2 Pressure	SR 3.5.1.3	4.5.1.a.1)
Verify Boron Concentration	SR 3.5.1.4	4.5.1.b
Verify Power removed from isolation valve	SR 3.5.1.5	4.5.1.c
ECCS – Operating		
Verify Valve Lineup	SR 3.5.2.1	4.5.2.a
Verify Valve Position	SR 3.5.2.2	4.5.2.b.2)
Verify Piping Sufficiently Full	SR 3.5.2.3	4.5.2.b.1)
Verify Automatic Valve Actuation	SR 3.5.2.5	4.5.2.e.1)

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Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
Verify Automatic Pump Start	SR 3.5.2.6	4.5.2.e.2)
Verify RHR Pump Stop Automatically – LoLo RWST	---	4.5.2.e.3)
Verify Throttle Valve Position	SR 3.5.2.7	4.5.2.g.2)
Inspection Sump Components	SR 3.5.2.8	4.5.2.d.2)
Verify Auto Interlock Prevents Valve Opening	---	4.5.2.d.1)
RWST		
Verify Water Temperature	SR 3.5.4.1	4.5.4.b & 4.6.2.1.a.2)
Verify Water Volume	SR 3.5.4.2	4.5.4.a.1)
Verify Boron Concentration	SR 3.5.4.3	4.5.4.a.2)
Seal Injection Flow		
Verify Throttle Valve Position	SR 3.5.5.1	---
pH Tri-sodium Phosphate Storage Baskets		
Verify Minimum volume of TSP	---	4.5.5
BIT		
Verify Water Temperature	SR 3.5.6.1	Note 1
Verify Water Volume	SR 3.5.6.2	Note 1
Verify Water Boron Concentration	SR 3.5.6.3	Note 1
Containment Air Locks		
Verify Interlock Operation	SR 3.6.2.2	4.6.1.3.c
Containment Isolation Valves		
Verify 42" Purge Valves Sealed Closed	SR 3.6.3.1	4.6.1.7.1
Verify 8" Purge Valves Closed	SR 3.6.3.2	4.6.1.7.1
Verify Valves Outside Containment in Correct Position	SR 3.6.3.3	4.6.1.1.a
Verify Isolation Time of Valves	SR 3.6.3.5	---
Cycle Weight/Spring Loaded Check Valves	SR 3.6.3.6	---
Perform Leak Rate Test of Purge Valves	SR 3.6.3.7	---
Verify Automatic Valves Actuate to Correct Position	SR 3.6.3.8	4.6.3.2.a, b & c
Cycle Non Testable Weight/Spring Loaded Check Valves	SR 3.6.3.9	---
Verify Purge Valves Blocked	SR 3.6.3.10	---
Containment Pressure		
Verify Pressure	SR 3.6.4.1	4.6.1.4
Containment Air Temperature		
Verify Average Air Temperature	SR 3.6.5.1	4.6.1.5
Spray Systems		
Verify Valve Position	SR 3.6.6D.1	4.6.2.1.a.1)
Verify Valve Actuation	SR 3.6.6D.3	4.6.2.1.c.1)
Verify Pump Start on Auto Signal	SR 3.6.6D.4	4.6.2.1.c.2)
Verify Nozzle are not Obstructed	SR 3.6.6D.5	---

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Italicized text denotes MPS3-specific surveillances

Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
Recirculation Spray		
Verify Casing Cooling Temperature	SR 3.6.6E.1	---
Verifying Casing Cooling Volume	SR 3.6.6E.2	---
Verify Casing Cooling Boron Concentration	SR 3.6.6E.3	----
Verify Valve Position	SR 3.6.6E.4	4.6.2.2.a
Verify Actuation of Pumps and Valves	SR 3.6.6E.6	4.6.2.2.c & d
Verify Nozzle are not Obstructed	SR 3.6.6E.7	---
Spray Additive System		
Verify Valve position	SR 3.6.7.1	Note 1
Verify Tank Volume	SR 3.6.7.2	Note 1
Verify Tank Solution Concentration	SR 3.6.7.3	Note 1
Actuate Each Flow Path Valve	SR 3.6.7.4	Note 1
Verify Spray Additive Flow Rate	SR 3.6.7.5	Note 1
Iodine Cleanup System		
Operate train with heaters	SR 3.6.11.1	Note 1
Verify train Actuation	SR 3.6.11.3	Note 1
Verify Filter Bypass Operation	SR 3.6.11.4	Note 1
Steam Jet Air Ejectors		
Verify Air Ejector outside Containment Isolation Valve Closed	---	4.6.5.1.1
Supplementary Leak Collection and Release System MPS3		
Verify Manual Train Actuation & Operate Heaters	SR 3.6.13.1	4.6.6.1.a
Verify Filter Penetration and Bypass Leakage	---	4.6.6.1.b.1)
Verify Filter Pressure Drop	---	4.6.6.1.d.1)
Verify Acutation on Safety Injection Signal	SR 3.6.13.3	4.6.6.1.d.2)
Verify Heater Capacity	---	4.6.6.1.d.3)
Verify Damper Open	SR 3.6.13.4	---
Verify Each train Flow	SR 3.6.13.5	4.6.6.1.a./4.6.6.1.b.3)
Secondary Containment (MPS3)		
Verify Doors in Access Opening are Closed	---	4.6.6.2.1
Verify Each SLCR System Produce Negative Pressure	---	4.6.6.2.2
Main Steam Isolation Valves		
Actuate Valves	SR 3.7.2.2	---
MFIVs and MFRVs		
Actuate Valves	SR 3.7.3.2	---
Atmospheric Dump Valves –		
Cycle Dump Valves	SR 3.7.4.1	4.7.1.6.1
Cycle Block Valves	SR 3.7.4.2	4.7.1.6.2

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Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
AFW		
Verify Valve Position	SR 3.7.5.1	4.7.1.2.1.a
Verify Auto Valve Actuation	SR 3.7.5.3	---
Verify Pump Auto Actuation	SR 3.7.5.4	4.7.1.2.1.c
Verify Pump Head	---	4.7.1.2.1.b.1 & 2
Condensate Storage Tank		
Verify Volume of DWST	SR 3.7.6.1	4.7.1.3.1
Component Cooling		
Verify Valve Position	SR 3.7.7.1	4.7.3.a
Verify Valve Actuation	SR 3.7.7.2	4.7.3.b.1)
Verify Pump Actuation	SR 3.7.7.3	4.7.3.b.2)
Service Water		
Verify Valve Position	SR 3.7.8.1	4.7.4.a
Verify Valve Actuation	SR 3.7.8.2	4.7.4.b.1)
Verify Pump Actuation	SR 3.7.8.3	4.7.4.b.2)
Ultimate Heat Sink		
Verify Water Level	SR 3.7.9.1	---
Verify Water Temperature	SR 3.7.9.2	4.7.5.a
Operate Cooling Tower	SR 3.7.9.3	Note 1
Verify Fan Actuation	SR 3.7.9.4	Note 1
CR Emergency Ventilation		
Verify Control Room Air Temperature	---	4.7.7.a
Verify Manual Train Actuation Operate Heaters	SR 3.7.10.1	4.7.7.b
Verify Train Actuation Actual or Simulated Signal	SR 3.7.10.3	---
Verify Envelope Pressurization	SR 3.7.10.4	---
Verify Filter Penetration and Bypass Leakage	---	4.7.7.c.1)
Verify System Flow Rate	---	4.7.7.b/4.7.7.c.3)
Verify Filter Pressure Drop	---	4.7.7.e.1)
Verify Heater Capacity	---	4.7.7.e.3)
CR Air Condition System		
Verify Train Capacity	SR 3.7.11.1	---
ECCS PREACS *(MPS3 Auxiliary Building Filter System)*		
Verify Manual Train Actuation and Operate Heaters	SR 3.7.12.1	4.7.9.a
Verify Train Actuation Actual or Simulated Signal	SR 3.7.12.3	4.7.9.d.2)
Verify Filter Penetration and Bypass Leakage	---	4.7.9.b.1)
Verify System Flow Rate	---	4.7.9.a/4.7.9.b.3)
Verify Envelope Negative Pressure	SR 3.7.12.4	---
Verify Pressure Drop Across HEPA and Adsorbers	---	4.7.9.d.1)

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Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
Verify Bypass Damper Closure	SR 3.7.12.5	---
Verify Heater Capacity	---	4.7.9.d.3)
Fuel Building Air Cleanup		
Operate Heaters	SR 3.7.13.1	---
Verify Automatic Train Actuation	SR 3.7.13.3	---
Verify Envelope Negative Pressure	SR 3.7.13.4	---
Verify Bypass Damper Closure	SR 3.7.13.5	---
Penetration Room Air Cleanup System –		
Operate Heaters	SR 3.7.14.1	Note 1
Verify Automatic Train Actuation	SR 3.7.14.3	Note 1
Verify Envelope Negative Pressurization	SR 3.7.14.4	Note 1
Verify Bypass Damper Closure	SR 3.7.14.5	Note 1
Fuel Storage Pool Water Level		
Verify Water Level	SR 3.7.15.1	4.9.11
Fuel Storage Pool Boron		
Verify Boron Concentration	SR 3.7.16.1	4.9.1.2
Secondary Specific Activity		
Verify Secondary Activity	SR 3.7.18.1	4.7.1.4
Area Temperature Monitoring (MPS3)		
Verify Temperature is Within Limits	---	4.7.14.a
AC Sources –Operating		
Verify Breaker Alignment Offsite Circuits	SR 3.8.1.1	4.8.1.1.1.a
Verify EDG Starts - Achieves Voltage & Frequency	SR 3.8.1.2	4.8.1.1.2.a.5)
Synchronize and Load for > 60 minutes	SR 3.8.1.3	4.8.1.1.2.a.6) & 4.8.1.1.2.b.2)
Verify Day Tank Level	SR 3.8.1.4	4.8.1.1.2.a.1)
Remove Accumulate Water for Day Tank	SR 3.8.1.5	4.8.1.1.2.c
Verify Operation of Transfer Pump	SR 3.8.1.6	4.8.1.1.2.a.3)
Verify EDG Starts – Achieves Voltage & Frequency in 10 seconds	SR 3.8.1.7	4.8.1.1.2.b.1)
Verify Auto and Manual Transfer of AC power Sources – Offsite Sources	SR 3.8.1.8	4.8.1.1.1.b
Verify the EDG alignment for standby power	---	4.8.1.1.2.a.7)
Verify Largest Load Rejection	SR 3.8.1.9	4.8.1.1.2.g.2)
Verify EDG Does Not Trip with Load Rejection	SR 3.8.1.10	4.8.1.1.2.g.3)
Verify De-energize, Load Shed and Re energize Emergency Bus with Loss of Offsite Power	SR 3.8.1.11	4.8.1.1.2.g.4)a) & b)
Verify EDG Start on ESF Signal	SR 3.8.1.12	4.8.1.1.2.g.5)

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Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
Verify EDG Noncritical Trips are Bypassed	SR 3.8.1.13	4.8.1.1.2.g.6)c)
Verify the Lockout Features Prevent EDG Starting	---	4.8.1.1.2.l
Run EDG for 24 Hours	SR 3.8.1.14	4.8.1.1.2.j
Verify EDG Starts Post Operation – Achieves Voltage & Frequency	SR 3.8.1.15	---
Verify EDG Synchronizes w/ Offsite Power and Transfers Load	SR 3.8.1.16	4.8.1.1.2.g.9) a), b) &c)
Verify ESF Signal overrides Test Mode of EDG	SR 3.8.1.17	4.8.1.1.2.g.10)
Verify Load Sequencers are with Design Tolerance	SR 3.8.1.18	4.8.1.1.2.g.12)
Verify EDG Start on Loss of Offsite Power with ESF	SR 3.8.1.19	4.8.1.1.2.g.6)a) & b)
Verify When Started Simultaneously from Standby Each EDGs Reach Rated Voltage and Frequency	SR 3.8.1.20	4.8.1.1.2.h
Verify auto-connected loads are < 5335kW	---	4.8.1.1.2.g.8)
Diesel FO and Starting Air		
Verify FO Storage Tank Volume	SR 3.8.3.1	4.8.1.1.2.a.2)
Verify Lube Oil Inventory	SR 3.8.3.2	4.8.1.1.2.a.4)
Verify EDG Air Start Receive Pressure	SR 3.8.3.4	---
Check and Remove Accumulate Water from FO Tanks	SR 3.8.3.5	4.8.1.1.2.d
Verify Total Particulate ≤ 10mg/liter	---	4.8.1.1.2.f
Verify Operation of FO Transfer Pumps via the Installed Cross-Connect Lines	---	4.8.1.1.2.k
Clean FO Storage Tanks	---	4.8.1.1.2.i
DC Sources Operating		
Verify Battery Terminal Voltage	SR 3.8.4.1	4.8.2.1.a.2)
Verify Station Battery Chargers Capable of Supplying [x]Amp for [y]Hours	SR 3.8.4.2	4.8.2.1.c.4)
Verify Battery Capacity	SR 3.8.4.3	4.8.2.1.d
Verify No Visable Corrosion at Terminal or Connectors and resistance > xx ohms	---	4.8.2.1.b.2)
Verify No Visual Indication of Physical Damage	---	4.8.2.1.c.1)
Cell-to-Cell And Terminal Connections Clean & Tight	---	4.8.2.1.c.2)
Verify Cell-to-Cell Resistance is ≤ xx ohms	---	4.8.2.1.c.3)
Battery Parameters		
Verify each Battery Float Current is ≤ [2] amps.	SR 3.8.6.1	4.8.2.1.a.1 & b.1)
Verify each Battery Pilot Cell Voltage is ≥[2.07] V	SR 3.8.6.2	4.8.2.1.a.1 & b.1)
Verify each Battery Cell Electrolyte Level is ≥ to Minimum Design Limits.	SR 3.8.6.3	4.8.2.1.a.1 & b.1)
Verify each Battery Pilot Cell Temperature ≥ to Minimum Design Limits.	SR 3.8.6.4	4.8.2.1.b.3)

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Technical Specification Section Title/ Surveillance Description*	TSTF 425	MPS3
Verify Each Battery Connected Cell Voltage is≥[2.07] V.	SR 3.8.6.5	4.8.2.1.b.1)
Verify Station and EDG Battery Capacity - >80% After Performance Test	SR 3.8.6.6	4.8.2.1.e
Complete Performance Discharge Test if Signs of Degradation or after reaching 85% of Service Life	---	4.8.2.1.f
Inverters - Operating		
Verify Correct Inverter Voltage & Alignment to Required AC Vital Buses.	SR 3.8.7.1	---
Inverters - Shutdown		
Verify Correct Inverter Voltage & Alignment to Required AC Vital Buses.	SR 3.8.8.1	---
Distribution System - Operating		
Verify Correct Breaker Alignments & Voltage to AC/DC and AC Vital Bus Electrical Distribution Subsystems.	SR 3.8.9.1	4.8.3.1
Distribution System - Shutdown		
Verify Correct Breaker Alignments & Voltage to AC/DC and AC Vital Bus Electrical Distribution Subsystems.	SR 3.8.10.1	4.8.3.2
Boron Concentration		
Verify Boron Concentration is Within COLR Limit	SR 3.9.1.1	4.9.1.1.2
Primary Grade Water Source Isolation Valves		
Verify Each Valve that Isolates Unborated Water Sources is Secured in the Closed Position	SR 3.9.2.1	4.9.1.1.3
Nuclear Instrumentation		
Perform Channel Check	SR 3.9.3.1	4.9.2.a
Perform Channel Calibration	SR 3.9.3.2	4.9.2.b
Containment Penetrations		
Verify each Required Containment Penetration is in the Required Status.	SR 3.9.4.1	4.9.4.a
Verify Each Required Containment Purge and Exhaust Valve Actuates to the Isolation Position on an Actuated or Simulated Actuation Signal.	SR 3.9.4.2	---
RHR and Coolant Circulation - *High Water Level*		
Verify One Loop is in Operation and Circulating Reactor Coolant at a Flow Rate of > [2800] gpm.	SR 3.9.5.1	4.9.8.1
RHR and Coolant Circulation - *Low Water Level*		
Verify One Loop is in Operation and Circulating Reactor Coolant at a flow rate of > [2800] gpm.	SR 3.9.6.1	4.9.8.2
Verify Correct Breaker Alignment and Indicated Power Available to Required RHR Pump Not in Operation.	SR 3.9.6.2	---
Refueling Cavity Water Level		
Verify Refueling Cavity Water Level is ≥23 ft Above The Top of Reactor Vessel Flange.	SR 3.9.7.1	4.9.10

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ATTACHMENT 5

SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION

Description of Amendment Request:

This amendment request involves the adoption of approved changes to the standard technical specifications (STS) for Westinghouse Pressurized Water Reactors (NUREG-1431), to allow relocation of specific technical specification (TS) surveillance frequencies to a licensee controlled program. The proposed changes are described in Technical Specification Task Force (TSTF) Traveler, TSTF-425, Revision 3 (ADAMS Accession No. ML090850642), "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b" and are described in the Notice of Availability published in the Federal Register on July 6, 2009 (74 FR 31996).

The proposed changes are consistent with NRC-approved Industry/TSTF Traveler, TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control-RITSTF Initiative 5b." The proposed changes relocate surveillance frequencies to a licensee controlled program, the Surveillance Frequency Control Program (SFCP). The changes are applicable to licensees using probabilistic risk guidelines contained in NRC-approved NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," (ADAMS Accession No. 071360456). In addition, administrative/editorial deviations of the TSTF-425 inserts and the existing TS wording are being proposed to fit the custom TS format.

Basis for proposed no significant hazards consideration: As required by 10 CFR 50.91 (a), the Dominion analysis of the issue of no significant hazards consideration is presented below:

1. Do the proposed changes involve a significant increase in the probability or consequences of any accident previously evaluated?

Response: No.

The proposed changes relocate the specified frequencies for periodic surveillance requirements to licensee control under a new Surveillance Frequency Control Program. Surveillance frequencies are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the TSs for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any previously evaluated?

Response: No.

No new or different accidents result from utilizing the proposed changes. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in the margin of safety?

Response: No.

The design, operation, testing methods, and acceptance criteria for systems, structures, and components (SSCs), specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant licensing basis (including the final safety analysis report and bases to TS), since these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. To evaluate a change in the relocated surveillance frequency, Dominion will perform a probabilistic risk evaluation using the guidance contained in NRC approved NEI 04-10, Rev. 1, in accordance with the TS SFCP. NEI 04-10, Rev. 1, methodology provides reasonable acceptance guidelines and methods for evaluating the risk increase of proposed changes to surveillance frequencies consistent with Regulatory Guide 1.177.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above, Dominion concludes that the requested changes do not involve a significant hazards consideration as set forth in 10 CFR 50.92(c), Issuance of Amendment.

ATTACHMENT 6

MARKED-UP TECHNICAL SPECIFICATIONS BASES CHANGES

(For Information Only)

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

BASES FOR
SECTIONS 3.0 AND 4.0
LIMITING CONDITIONS FOR OPERATION
AND
SURVEILLANCE REQUIREMENTS

Insert 2
The surveillance frequency is controlled under the Surveillance Frequency Control Program.

at frequency specified in the Surveillance Frequency Control Program

MOVABLE CONTROL ASSEMBLIES (Continued)

Control rod positions and OPERABILITY of the rod position indicators are required to be verified ~~on a nominal basis of once per 12 hours~~ with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The Digital Rod Position Indication (DRPI) System is defined as follows:

- Rod position indication as displayed on DRPI display panel (MB4), or
- Rod position indication as displayed by the Plant Process Computer System.

With the above definition, LCO 3.1.3.2, "ACTION a." is not applicable with either DRPI display panel or the plant process computer points OPERABLE.

The plant process computer may be utilized to satisfy DRPI System requirements which meets LCO 3.1.3.2, in requiring diversity for determining digital rod position indication.

Technical Specification SR 4.1.3.2.1 determines each digital rod position indicator to be OPERABLE by verifying the Demand Position Indication System and the DRPI System agree within 12 steps at ~~least once each 12 hours~~, except during the time when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the DRPI System at least once each 4 hours.

the frequency specified in the Surveillance Frequency Control Program

The Rod Deviation Monitor is generated only from the DRPI panel at MB4. Therefore, when rod position indication as displayed by the plant process computer is the only available indication, then perform SURVEILLANCE REQUIREMENTS every 4 hours.

~~Technical Specification SR 4.1.3.2.1 determines each digital rod position indicator to be OPERABLE by verifying the Demand Position Indication System and the DRPI System agree within 12 steps at least once each 12 hours, except during the time when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the DRPI System at least once each 4 hours.~~

~~The Rod Deviation Monitor is generated only from the DRPI panel at MB4. Therefore, when rod position indication as displayed by the plant process computer is the only available indication, then perform SURVEILLANCE REQUIREMENTS every 4 hours.~~

Duplicative paragraphs

the frequency specified in the Surveillance Frequency Control Program

MOVABLE CONTROL ASSEMBLIES (Continued)

Additional surveillance is required to ensure the plant process computer indications are in agreement with those displayed on the DRPI. This additional SURVEILLANCE REQUIREMENT is as follows:

> Each rod position indication as displayed by the plant process computer shall be determined to be OPERABLE by verifying the rod position indication as displayed on the DRPI display panel agrees with the rod position indication as displayed by the plant process computer at ~~least once per 12 hours~~.

The rod position indication, as displayed by DRPI display panel (MB4), is a non-QA system, calibrated on a refueling interval, and used to implement T/S 3.1.3.2. Because the plant process computer receives field data from the same source as the DRPI System (MB4), and is also calibrated on a refueling interval, it fully meets all requirements specified in T/S 3.1.3.2 for rod position. Additionally, the plant process computer provides the same type and level of accuracy as the DRPI System (MB4). The plant process computer does not provide any alarm or rod position deviation monitoring as does DRPI display panel (MB4).

For Specification 3.1.3.1 ACTIONS b. and c., it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). Trippability is defined in Attachment C to a letter dated December 21, 1984, from E. P. Rahe (Westinghouse) to C. O. Thomas (NRC). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus falls under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately 4 hours for this verification.

For LCO 3.1.3.6 the control bank insertion limits are specified in the CORE OPERATING LIMITS REPORT (COLR). These insertion limits are the initial assumptions in safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions, assumptions of available SHUTDOWN MARGIN, and initial reactivity insertion rate.

The applicable I&C calibration procedure (Reference 1.) being current indicates the associated circuitry is OPERABLE.

There are conditions when the Lo-Lo and Lo alarms of the RIL Monitor are limited below the RIL specified in the COLR. The RIL Monitor remains OPERABLE because the lead control rod bank still has the Lo and Lo-Lo alarms greater than or equal to the RIL.

in accordance with the Surveillance Frequency Control Program

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset the effect of rod bow and any other DNB penalties that may occur. The remaining margin is available for plant design flexibility.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The heat flux hot channel factor, $F_Q(Z)$, is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions. Using the measured three dimensional power distributions, it is possible to derive $F_Q^M(Z)$, a computed value of $F_Q(Z)$. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations.

To account for these possible variations, the steady state limit of $F_Q(Z)$ is adjusted by an elevation dependent factor appropriate to either RAOC or base load operation, W(Z) or $W(Z)_{BL}$, that accounts for the calculated worst case transient conditions. The W(Z) and $W(Z)_{BL}$, factors described above for normal operation are specified in the COLR per Specification 6.9.1.6. Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion. Evaluation of the steady state $F_Q(Z)$ limit is performed in Specification 4.2.2.1.2.b and 4.2.2.1.4.b while evaluation nonequilibrium limits are performed in Specification 4.2.2.1.2.c and 4.2.2.1.4.c.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of the Limiting Condition for Operation. Measurement errors for RCS total flow rate and for $F^N_{\Delta H}$ have been taken into account in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. To perform the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated ~~at least once per 18 months~~. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Any fouling which might bias the RCS flow rate measurement can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

in accordance with the Surveillance Frequency Control Program

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The ~~12-hour~~ periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation defined in Specifications 3.2.3.1.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during POWER OPERATION.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the design limit throughout each analyzed transient. The indicated T_{avg} values

DNB PARAMETERS (Continued)

and the indicated pressurizer pressure values are specified in the CORE OPERATING LIMITS REPORT. The calculated values of the DNB related parameters will be an average of the indicated values for the OPERABLE channels.

The ~~12-hour~~ periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. Measurement uncertainties have been accounted for in determining the parameter limits.

in accordance with the Surveillance Frequency Control Program

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated action and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed ~~at the minimum frequencies~~ are sufficient to demonstrate this capability. Insert 2

The Engineered Safety Features Actuation System Nominal Trip Setpoints specified in Table 3.3-4 are the nominal values of which the bistables are set for each functional unit. The Allowable Values (Nominal Trip Setpoints ± the calibration tolerance) are considered the Limiting Safety System Settings as identified in 10CFR50.36 and have been selected to mitigate the consequences of accidents. A Setpoint is considered to be consistent with the nominal value when the measured "as left" Setpoint is within the administratively controlled (±) calibration tolerance identified in plant procedures (which specifies the difference between the Allowable Value and Nominal Trip Setpoint). Additionally, the Nominal Trip Setpoints may be adjusted in the conservative direction provided the calibration tolerance remains unchanged.

Measurement and Test Equipment accuracy is administratively controlled by plant procedures and is included in the plant uncertainty calculations as defined in WCAP-10991. OPERABILITY determinations are based on the use of Measurement and Test Equipment that conforms with the accuracy used in the plant uncertainty calculation.

The Allowable Value specified in Table 3.3-4 defines the limit beyond which a channel is inoperable. If the process rack bistable setting is measured within the "as left" calibration tolerance, which specifies the difference between the Allowable Value and Nominal Trip Setpoint, then the channel is considered to be OPERABLE.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The methodology, as defined in WCAP-10991 to derive the Nominal Trip Setpoints, is based upon combining all of the uncertainties in the channels. Inherent in the determination of the Nominal Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels should be capable of operating within the allowances of these uncertainty magnitudes. Occasional drift in excess of the allowance may be determined to be acceptable based on the other device performance characteristics. Device drift in excess of the allowance that is more than occasional, may be indicative of more serious problems and would warrant further investigation.

The above Bases does not apply to the Control Building Inlet Ventilation radiation monitors ESF Table (Item 7E). For these radiation monitors the allowable values are essentially nominal values. Due to the uncertainties involved in radiological parameters, the methodologies of WCAP-10991 were not applied. Actual trip setpoints will be reestablished below the allowable value based on calibration accuracies and good practices.

The OPERABILITY requirements for Table 3.3-3, Functional Units 7.a, "Control Building Isolation, Manual Actuation," and 7.e, "Control Building Isolation, Control Building Inlet Ventilation Radiation," are defined by table notation "*". These functional units are required to be OPERABLE at all times during plant operation in MODES 1, 2, 3, and 4. These functional units are also required to be OPERABLE during movement of recently irradiated fuel assemblies, as specified by table notation "*". The Control Building Isolation Manual Actuation and Control Building Inlet Ventilation Radiation are required to be OPERABLE during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 350 hours*). Table notation "*" of Table 4.3-2 has the same applicability.

The verification of response time ~~at the specified frequencies~~ provides assurance that the reactor trip and the engineered safety features actuation associated with each channel is completed within the time limit assumed in the safety analysis. No credit is taken in the analysis for those channels with response times indicated as not applicable (i.e., N.A.). Insert 2

Required ACTION 4. of Table 3.3-1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this ACTION provided they are accounted for in the calculated SDM. The proposed change permits operations introducing positive reactivity additions but prohibits the temperature change or overall boron concentration from decreasing below that required to maintain the specified SDM or required boron concentration.

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* During fuel assembly cleaning evolutions that involve the handling or cleaning of two fuel assemblies coincidentally, recently irradiated fuel is fuel that has occupied part of a critical reactor core within the previous 525 hours.

Required ACTION b. is modified by a Note which permits plant temperature changes provided the temperature change is accounted for in the calculated SDM. Introduction of temperature changes, including temperature increases when a positive MTC exists, must be evaluated to ensure they do not result in a loss of required SDM.

2. All dilution flowpaths are isolated and placed under administrative control (locked closed). This action provides redundant protection and defense in depth (safety overlap) to the SMMs. In this configuration, a boron dilution event (BDE) cannot occur. This is the basis for not having to analyze for BDE in MODE 6. Since the BDE cannot occur with the dilution flow paths isolated, the SMMs are not required to be OPERABLE as the event cannot occur and OPERABLE SMMs provide no benefit.

3. Increase the SHUTDOWN MARGIN surveillance frequency from ~~every 24 hours~~ to every 12 hours. This action in combination with the above, provide defense in depth and overlap to the loss of the SMMs.

the frequency specified in the Surveillance Frequency Control Program

Surveillance Requirements

The SMMs are subject to an ~~ACOT~~ ~~every 92 days~~ to ensure each train of SMM is fully operational. This test shall include verification that the SMMs are set per the CORE OPERATING LIMITS REPORT. Insert 2

ANALOG CHANNEL OPERATIONAL TEST

3/4.4 REACTOR COOLANT SYSTEM

BASES

The safety analyses performed for the reactor at power assume that all reactor coolant loops are initially in operation and the loop stop valves are open. This LCO places controls on the loop stop valves to ensure that the valves are not inadvertently closed in MODES 1, 2, 3 and 4. The inadvertent closure of a loop stop valve when the Reactor Coolant Pumps (RCPs) are operating will result in a partial loss of forced reactor coolant flow. If the reactor is at rated power at the time of the event, the effect of the partial loss of forced coolant flow is a rapid increase in the coolant temperature which could result in DNB with subsequent fuel damage if the reactor is not tripped by the Low Flow reactor trip. If the reactor is shutdown and a RCS loop is in operation removing decay heat, closure of the loop stop valve associated with the operating loop could also result in increasing coolant temperature and the possibility of fuel damage.

The loop stop valves have motor operators. If power is inadvertently restored to one or more loop stop valve operators, the potential exists for accidental closure of the affected loop stop valve(s) and the partial loss of forced reactor coolant flow. With power applied to a valve operator, only the interlocks prevent the valve from being operated. Although operating procedures and interlocks make the occurrence of this event unlikely, the prudent action is to remove power from the loop stop valve operators. The time period of 30 minutes to remove power from the loop stop valve operators is sufficient considering the complexity of the task.

Should a loop stop valve be closed in MODES 1 through 4, the affected valve must be maintained closed and the plant placed in MODE 5. Once in MODE 5, the isolated loop may be started in a controlled manner in accordance with LCO 3.4.1.6, "Reactor Coolant System Isolated Loop Startup." Opening the closed loop stop valve in MODES 1 through 4 could result in colder water or water at a lower boron concentration being mixed with the operating RCS loops resulting in positive reactivity insertion. The time period provided in ACTION 3.4.1.5.b allows time for borating the operating loops to a shutdown boration level such that the plant can be brought to MODE 3 within 6 hours and MODE 5 within 30 hours. The allowed ACTION times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Surveillance Requirement 4.4.1.5 is performed ~~at least once per 31 days~~ to ensure that the RCS loop stop valves are open, with power removed from the loop stop valve operators. The primary function of this Surveillance is to ensure that power is removed from the valve operators, since Surveillance Requirement 4.4.1.1 requires verification ~~every 12 hours~~ that all loops are operating and circulating reactor coolant, thereby ensuring that the loop stop valves are open. The frequency ~~of 31 days~~ ensures that the required flow is available~~, is based on engineering judgement, and has proven to be acceptable. Operating experience has shown that the failure rate is so low that the 31 day frequency is justified.~~

Insert 2

specified in the Surveillance Frequency Control Program

- For the isolated loop being restored, the power to both loop stop valves has been restored

Surveillance 4.4.1.6.2 indicates that the reactor shall be determined subcritical by at least the amount required by Specifications 3.1.1.1.2 or 3.1.1.2 for MODE 5 or Specification 3.9.1.1 for MODE 6 within 2 hours of opening the cold leg or hot leg stop valve.

The SHUTDOWN MARGIN requirement in Specification 3.1.1.1.2 is specified in the CORE OPERATING LIMITS REPORT for MODE 5 with RCS loops filled. Specification 3.1.1.1.2 cannot be used to determine the required SHUTDOWN MARGIN for MODE 5 loops isolated condition.

Specification 3.1.1.2 requires the SHUTDOWN MARGIN to be greater than or equal to the limits specified in the CORE OPERATING LIMITS REPORT for MODE 5 with RCS loops not filled provided CVCS is aligned to preclude boron dilution. This specification is for loops not filled and therefore is applicable to an all loops isolated condition.

Specification 3.9.1.1 requires K_{eff} of 0.95 or less, or a boron concentration of greater than or equal to the limit specified in the COLR in MODE 6.

Specification 3.1.1.1.2 or 3.1.1.2 for MODE 5, both require boron concentration to be determined at ~~least once each 24 hours~~. SR 4.1.1.1.2.1.b.2 and 4.1.1.2.1.b.1 satisfy the requirements of Specifications 3.1.1.1.2 and 3.1.1.2 respectfully. Specification 3.9.1.1 for MODE 6 requires boron concentration to be determined at ~~least once each 72 hours~~. S.R. 4.9.1.1.2 satisfy the requirements of Specification 3.9.1.1.

the frequency specified in the Surveillance Frequency Control Program

Per Specifications 3.4.1.2, ACTION c.; 3.4.1.3, ACTION c.; 3.4.1.4.1, ACTION b.; and 3.4.1.4.2, ACTION b., suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1.1.2 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however, coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations.

References:

1. Letter NEU-94-623, dated July 13, 1994; Mixing Evaluation for Boron Dilution Accident in Modes 4 and 5, Westinghouse HR-59782.

2. Memo No. MP3-E-93-821, dated October 7, 1993.

3/4.4.3 PRESSURIZER (continued)

Insert 2

The ~~12-hour~~ periodic surveillances require that pressurizer level be maintained at programmed level within ± 6% of full scale. The surveillance is performed by observing the indicated level. ~~The 12-hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and to ensure that the appropriate level exists in the pressurizer.~~ During transitory conditions, i.e., power changes, the operators will maintain programmed level, and deviations greater than 6% will be corrected within 2 hours. Two hours has been selected for pressurizer level restoration after a transient to avoid an unnecessary downpower with pressurizer level outside the operating band. Normally, alarms are also available for early detection of abnormal level indications.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of the reactor coolant. Unless adequate heater capacity is available, the hot high-pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single-phase natural circulation and decreased capability to remove core decay heat.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity of at least 175 kW. The heaters are capable of being powered from either the offsite power source or the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The requirement for two groups of pressurizer heaters, each having a capacity of 175 kW, is met by verifying the capacity of the pressurizer heater groups A and B. Since the pressurizer heater groups A and B are supplied from the emergency 480V electrical buses, there is reasonable assurance that these heaters can be energized during a loss of offsite power to maintain natural circulation at HOT STANDBY. Providing an emergency (Class 1E) power source for the required pressurizer heaters meets the requirement of NUREG-0737, "A Clarification of TMI Action Plan Requirements," II.E.3.1, "Emergency Power Requirements for Pressurizer Heaters."

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering that a demand caused by loss of offsite power would be unlikely in this time period. Pressure control may be maintained during this time using normal station powered heaters.

MODE 3

The requirement for the pressurizer to be OPERABLE, with a level less than or equal to 89%, ensures that a steam bubble exists. The 89% level preserves the steam space for pressure control. The 89% level has been established to ensure the capability to establish and maintain pressure control for MODE 3 and to ensure a bubble is present in the pressurizer. Initial pressurizer level is not significant for those events analyzed for MODE 3 in Chapter 15 of the FSAR.

3/4.4.3 PRESSURIZER (cont'd.)

Insert 2

The ~~12-hour~~ periodic surveillance requires that during MODE 3 operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The surveillance is performed by observing the indicated level. ~~The 12-hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and to ensure that a steam bubble exists in the pressurizer.~~ Alarms are also available for early detection of abnormal level indications.

The basis for the pressurizer heater requirements is identical to MODES 1 and 2.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Requiring the PORVs to be OPERABLE ensures that the capability for depressurization during safety grade cold shutdown is met.

ACTION statements a, b, and c distinguishes the inoperability of the power operated relief valves (PORV). Specifically, a PORV may be designated inoperable but it may be able to automatically and manually open and close and therefore, able to perform its function. PORV inoperability may be due to seat leakage which does not prevent automatic or manual use and does not create the possibility for a small-break LOCA. For these reasons, the block valve may be closed but the action requires power to be maintained to the valve. This allows quick access to the PORV for pressure control. On the other hand if a PORV is inoperable and not capable of being automatically and manually cycled, it must be either restored or isolated by closing the associated block valve and removing power.

Note: PORV position indication does not affect the ability of the PORV to perform any of its safety functions. Therefore, the failure of PORV position indication does not cause the PORV to be inoperable. However, failed position indication of these valves must be restored "as soon as practicable" as required by Technical Specification 6.8.4.e.3.

Automatic operation of the PORVs is created to allow more time for operators to terminate an Inadvertent ECCS Actuation at Power. The PORVs and associated piping have been demonstrated to be qualified for water relief. Operation of the PORVs will prevent water relief from the pressurizer safety valves for which qualification for water relief has not been demonstrated. If the PORVs are capable of automatic operation but have been declared inoperable, closure of the PORV block valve is acceptable since the Emergency Operating Procedures provide guidance to assure that the PORVs would be available to mitigate the event. OPERABILITY and setpoint controls for the safety grade PORV opening logic are maintained in the Technical Requirements Manual.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in RCS LCO 3.4.6.1, "Leakage Detection Systems."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

~~The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.~~

Insert 2

4.4.6.2.1.e

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.5, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

Insert 2

~~The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.~~ The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Reference 5).

4.4.6.2.2

The Surveillance Requirements for RCS pressure isolation valves provide assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

SPECIFIC ACTIVITY (Continued)

ACTIONS (Continued)

e.

If the required action and completion time of ACTION d. is not met, the reactor must be brought to HOT STANDBY (MODE 3) within 6 hours and COLD SHUTDOWN (MODE 5) within 36 hours. The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

4.4.8.1

Surveillance Requirement 4.4.8.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at ~~least once every 7 days~~ the frequency specified in the Surveillance Frequency Control Program. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance Requirement provides an indication of any increase in the noble gas specific activity.

Trending the results of this Surveillance Requirement allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. ~~The surveillance 7 day frequency considers the low probability of a gross fuel failure during this time.~~

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the Surveillance Requirement 4.4.8.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity. Insert 2

A Note modifies the Surveillance Requirement to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the Surveillance Requirement. This allows the Surveillance Requirement to be performed in those MODES, prior to entering MODE 1.

4.4.8.2

This Surveillance Requirement is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. ~~The 14 day frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days.~~ The frequency of between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

OVERPRESSURE PROTECTION SYSTEMS (continued)

The AOT in MODE 4 considers the facts that only one of the relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low. The RCS must be depressurized and a vent must be established within the following 12 hours if the required relief valve is not restored to OPERABLE within the required AOT of 7 days.

d.

The consequences of operational events that will overpressure the RCS are more severe at lower temperatures (Ref. 8). Thus, with one of the two required relief valves inoperable in MODE 5 or in MODE 6 with the head on, the AOT to restore two valves to OPERABLE status is 24 hours.

The AOT represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE relief valve to protect against overpressure events. The RCS must be depressurized and a vent must be established within the following 12 hours if the required relief valve is not restored to OPERABLE within the required AOT of 24 hours.

e.

The RCS must be depressurized and a vent must be established within 12 hours when both required Cold Overpressure Protection relief valves are inoperable.

The vent must be sized ≥ 2.0 square inches to ensure that the flow capacity is greater than that required for the worst case cold overpressure transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible non-ductile failure of the reactor vessel.

The time required to place the plant in this Condition is based on the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1

Performance of an ANALOG CHANNEL OPERATIONAL TEST is required within 31 days prior to entering a condition in which the PORV is required to be OPERABLE and ~~every 31 days~~ on each required PORV to verify and, as necessary, adjust its lift setpoint. The ANALOG CHANNEL OPERATIONAL TEST will verify the setpoint in accordance with the nominal values given in Figures 3.4-4a and 3.4-4b. PORV actuation could depressurize the RCS; therefore, valve operation is not required.

at the frequency specified in the Surveillance Frequency Control Program thereafter

OVERPRESSURE PROTECTION SYSTEMS (~~continued~~)

periodically

Performance of a ~~CHANNEL~~ CALIBRATION on each required PORV actuation channel is required ~~once per 24 months~~ to adjust the channel so that it responds and the valve opens within the required range and accuracy to a known input.

The PORV block valve must be verified open and COPPS must be verified armed ~~every 72 hours~~ to provide a flow path and a cold overpressure protection actuation circuit for each required PORV to perform its function when required. The valve is remotely verified open in the main control room. This Surveillance is performed if credit is being taken for the PORV to satisfy the LCO.

Insert 2

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required to be removed, and the manual operator is not required to be locked in the open position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure transient.

~~The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify the PORV block valve remains open.~~

4.4.9.3.2

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying the RHR suction valves, 3RHS*MV8701A and 3RHS*M8701C, are open when suction relief valve 3RHS*RV8708A is being used to meet the LCO and by verifying the RHR suction valves, 3RHS*MV8702B and 3RHS*MV8702C, are open when suction relief valve 3RHS*RV8708B is being used to meet the LCO. Each required RHR suction relief valve shall also be demonstrated OPERABLE by testing it in accordance with 4.0.5. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

periodically

The RHR suction valves are verified to be open ~~every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valves remain open~~.

The ASME Code for Operation and Maintenance of Nuclear Power Plants, (Reference 9), test per 4.0.5 verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

OVERPRESSURE PROTECTION SYSTEMS (continued)

4.4.9.3.3

The RCS vent of ≥ 2.0 square inches is proven OPERABLE periodically by verifying its open condition. ~~either:~~

~~a. Once every 12 hours for a vent valve that cannot be locked open.~~

~~b. Once every 31 days for a valve that is locked, sealed, or secured in position or any other passive vent path.~~ A removed Pressurizer safety valve fits this category.

This passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO. Insert 2

4.4.9.3.4 and 4.4.9.3.5

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, all SIH pumps and all but one centrifugal charging pump are verified incapable of injecting into the RCS.

The SIH pumps and charging pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. Alternate methods of control may be employed using at least two independent means to prevent an injection into the RCS. This may be accomplished through any of the following methods: 1) placing the pump in pull to lock (PTL) and pulling its UC fuses, 2) placing the pump in pull to lock (PTL) and closing the pump discharge valve(s) to the injection line, 3) closing the pump discharge valve(s) to the injection line and either removing power from the valve operator(s) or locking manual valves closed, and 4) closing the valve(s) from the injection source and either removing power from the valve operator(s) or locking manual valves closed.

An SIH pump may be energized for testing or for filling the Accumulators provided it is incapable of injecting into the RCS.

~~The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.~~ Insert 2

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness for Protection Against Failure," 1995 Edition.
2. ASME Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," dated February 26, 1999.
3. Generic Letter 88-11
4. ASME, Boiler and Pressure Vessel Code, Section III
5. FSAR, Chapter 15
6. 10CFR50, Section 50.46
7. 10CFR50, Appendix K
8. Generic Letter 90-06
9. ASME Code for Operation and Maintenance of Nuclear Power Plants

ECCS SUBSYSTEMS (Continued)

flush upon heat exchanger return to service and procedural compliance is relied upon to ensure that gas is not present within the heat exchanger u-tubes.

Surveillance Requirement 4.5.2.C.2 requires that the visual inspection of the containment be performed at least once daily if the containment has been entered that day and when the final containment entry is made. This will reduce the number of unnecessary inspections and also reduce personnel exposure.

Insert 2

Surveillance Requirement 4.5.2.d.2 addresses periodic inspection of the containment sump to ensure that it is unrestricted and stays in proper operating condition. ~~The 24 month frequency is based on the need to perform this surveillance under the conditions that apply during an outage, and the need to have access to the location. This frequency is sufficient to detect abnormal degradation and is confirmed by operating experience.~~

The Emergency Core Cooling System (ECCS) has several piping cross connection points for use during the post-LOCA recirculation phase of operation. These cross-connection points allow the Recirculation Spray System (RSS) to supply water from the containment sump to the safety injection and charging pumps. The RSS has the capability to supply both Train A and B safety injection pumps and both Train A and B charging pumps. Operator action is required to position valves to establish flow from the containment sump through the RSS subsystems to the safety injection and charging pumps since the valves are not automatically repositioned. The quarterly stroke testing (Technical Specification 4.0.5) of the ECC/RSS recirculation flowpath valves discussed below will not result in subsystem inoperability (except due to other equipment manipulations to support valve testing) since these valves are manually aligned in accordance with the Emergency Operating Procedures (EOPs) to establish the recirculation flowpaths. It is expected the valves will be returned to the normal pre-test position following termination of the surveillance testing in response to the accident. Failure to restore any valve to the normal pre-test position will be indicated to the Control Room Operators when the ESF status panels are checked, as directed by the EOPs. The EOPs direct the Control Room Operators to check the ESF status panels early in the event to ensure proper equipment alignment. Sufficient time before the recirculation flowpath is required is expected to be available for operator action to position any valves that have not been restored to the pretest position, including local manual valve operation. Even if the valves are not restored to the pre-test position, sufficient capability will remain to meet ECCS post-LOCA recirculation requirements. As a result, stroke testing of the ECCS recirculation valves discussed below will not result in a loss of system independence or redundancy, and both ECCS subsystems will remain OPERABLE.

When performing the quarterly stroke test of 3SIH*MV8923A, the control switch for safety injection pump 3SIH*P1A is placed in the pull-to-lock position to prevent an automatic pump start with the suction valve closed. With the control switch for 3SIH*P1A in pull-to-lock, the Train A ECCS subsystem is inoperable and Technical Specification 3.5.2, ACTION a., applies. This ACTION statement is sufficient to administratively control the plant configuration with the automatic start of 3SIH*P1A defeated to allow stroke testing of 3SIH*MV8923A. In addition, the EOPs and the ESF status panels will identify this abnormal plant configuration, if not corrected following the termination of the surveillance testing, to the plant operators to allow restoration of the normal post-LOCA recirculation flowpath. Even if system restoration is not accomplished, sufficient equipment will be available to perform all ECCS and RSS injection and recirculation functions, provided no additional ECCS or RSS equipment is inoperable, and an additional single failure does not occur (an acceptable assumption since the Technical Specification ACTION statement limits the plant configuration time such that no additional equipment failure need be postulated). During the injection phase the redundant subsystem (Train B) is fully functional, as is a significant portion of the Train A subsystem. During the recirculation phase, the Train A RSS subsystem can supply water from the containment sump to the Train A

APPLICABILITY

In MODES 1, 2, 3, and 4, a design basis accident (DBA) could lead to a fission product release to containment that leaks to the secondary containment boundary. The large break LOCA, on which this system's design is based, is a full-power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and reactor coolant system pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequence of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the SLCRS is not required to be OPERABLE.

ACTIONS

If it is discovered that the TSP in the containment building sump is not within limits, action must be taken to restore the TSP to within limits. During plant operation, the containment sump is not accessible and corrections may not be possible.

The 7-day Completion Time is based on the low probability of a DBA occurring during this period. The Completion Time is adequate to restore the volume of TSP to within the technical specification limits.

If the TSP cannot be restored within limits within the 7-day Completion Time, the plant must be brought to a MODE in which the LCO does not apply. The specified Completion Times for reaching MODES 3 and 4 are those used throughout the technical specifications; they were chosen to allow reaching the specified conditions from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

Surveillance Requirement 4.5.5

Periodic determination of the volume of TSP in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the TSP during normal operation. ~~A Frequency of once per 24 months is required to determine visually that a minimum of 974 cubic feet is contained in the TSP Storage Baskets.~~ This requirement ensures that there is an adequate volume of TSP to adjust the pH of the post LOCA sump solution to a value ≥ 7.0. Insert 2

~~The periodic verification is required every refueling outage, since access to the TSP baskets is only feasible during outages. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.~~

The design of the Containment RSS is sufficiently independent so that an active failure in the recirculation spray mode, cold leg recirculation mode, or hot leg recirculation mode of the ECCS has no effect on its ability to perform its engineered safety function. In other words, the failure in one subsystem does not affect the capability of the other subsystem to perform its designated safety function of assuring adequate core cooling in the event of a design basis LOCA. As long as one subsystem is OPERABLE, with one pump capable of assuring core cooling and the other pump capable of removing heat from containment, the RSS system meets its design requirements.

The LCO 3.6.2.2. ACTION applies when any of the RSS pumps, heat exchangers, or associated components are declared inoperable. All four RSS pumps are required to be OPERABLE to meet the requirements of this LCO 3.6.2.2. During the injection phase of a Loss Of Coolant Accident all four RSS pumps would inject into containment to perform their containment heat removal function. The minimum requirement for the RSS to adequately perform this function is to have at least one subsystem available. Meeting the requirements of LCO 3.6.2.2. ensures the minimum RSS requirements are satisfied.

Surveillance Requirement 4.6.2.2.c requires that ~~at least once per 24 months,~~ verification is made that on a CDA test signal, each RSS pump starts automatically after receipt of an RWST Low-Low level signal. ~~The 24 month frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage and potential for unplanned transient if the surveillance was performed with the reactor at power. Operating experience has shown that these components pass the surveillance when performed at the 24 month frequency. Therefore the frequency was concluded to be acceptable from a reliability standpoint. This change has no adverse impact on plant safety.~~ Insert 2

Surveillance Requirements 4.6.2.1.d and 4.6.2.2.e require verification that each spray nozzle is unobstructed following maintenance that could cause nozzle blockage. Normal plant operation and maintenance activities are not expected to trigger performance of these surveillance requirements. However, activities, such as an inadvertent spray actuation that causes fluid flow through the nozzles, a major configuration change, or a loss of foreign material control when working within the respective system boundary may require surveillance performance. An evaluation, based on the specific situation, will determine the appropriate test method (e.g., visual inspection, air or smoke flow test) to verify no nozzle obstruction.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for these isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. FSAR Table 6.2-65 lists all containment isolation valves. The addition or deletion of any containment isolation valve shall be made in accordance with Section 50.59 of 10CFR50 and approved by the committee(s) as described in the QAP Topical Report.

For the purposes of meeting this LCO, the safety function of the containment isolation valves is to shut within the time limits assumed in the accident analyses. As long as the valves can shut within the time limits assumed in the accident analyses, the valves are OPERABLE. Where the valve position indication does not affect the operation of the valve, the indication is not required for valve OPERABILITY under this LCO. Position indication for containment isolation valves is covered by Technical Specification 6.8.4.e., Accident Monitoring Instrumentation. Failed position indication on these valves must be restored "as soon as practicable" as required by Technical Specification 6.8.4.e.3. Maintaining the valves OPERABLE, when position indication fails, facilitates troubleshooting and correction of the failure, allowing the indication to be restored "as soon as practicable."

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration.

If the containment isolation valve on a closed system becomes inoperable, the remaining barrier is a closed system since a closed system is an acceptable alternative to an automatic valve. However, actions must still be taken to meet Technical Specification ACTION 3.6.3.d and the valve, not normally considered as a containment isolation valve, and closest to the containment wall should be put into the closed position. No leak testing of the alternate valve is necessary to satisfy the ACTION statement. Placing the manual valve in the closed position sufficiently deactivates the penetration for Technical Specification compliance.

Closed system isolation valves applicable to Technical Specification ACTION 3.6.3.d are included in FSAR Table 6.2-65, and are the isolation valves for those penetrations credited as General Design Criteria 57. The specified time (i.e., 72 hours) of Technical Specification ACTION 3.6.3.d is reasonable, considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3 and 4. In the event the affected penetration is isolated in accordance with 3.6.3.d, the affected penetration flow path must be verified to be isolated on a periodic basis, (Surveillance Requirement 4.6.1.1.a). This is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. ~~The frequency of once per 31 days in this surveillance for verifying that each affected penetration flow path is isolated is appropriate considering the valves are operated under administrative controls and the probability of their misalignment is low.~~

Insert 2

3/4.6.6.1 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM (Continued)

Surveillance Requirements

Insert 2

a

Cumulative operation of the SLCRS with heaters operating for at least 10 continuous hours ~~in a 31-day period~~ is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. ~~The 31-day frequency was developed in consideration of the known reliability of fan motors and controls. This test is performed on a STAGGERED TEST BASIS once per 31-days.~~

b, c, e, and f

These surveillances verify that the required SLCRS filter testing is performed in accordance with Regulatory Guide 1.52, Revision 2. ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2. Laboratory testing of methyl iodide penetration shall be performed in accordance with ASTM D3803-89 and Millstone Unit 3 specific parameters. The surveillances include testing HEPA filter performance, charcoal adsorber efficiency, system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage.

Any time the OPERABILITY of a HEPA filter or charcoal adsorber housing has been affected by repair, maintenance, modification, or replacement activity, post maintenance testing in accordance with SR 4.0.1 is required to demonstrate OPERABILITY.

The 720 hours of operation requirement originates from Regulatory Guide 1.52, Revision 2, March 1978, Table 2, Note "c", which states that "Testing should be performed (1) initially, (2) at least once per 18 months thereafter for systems maintained in a standby status or after 720 hours of system operations, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system."

This testing ensures that the charcoal adsorbency capacity has not degraded below acceptable limits, as well as providing trend data. The 720 hour figure is an arbitrary number which is equivalent to a 30 day period. This criteria is directed to filter systems that are normally in operation and also provide emergency air cleaning functions in the event of a Design Basis Accident. The applicable filter units are not normally in operation and the sample canisters are typically removed due to the 18 month criteria.

d

periodic

Insert 2

The automatic startup ensures that each SLCRS train responds properly. The ~~once per 24 months frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance was performed with the reactor at power.~~ The surveillance verifies that the SLCRS starts on a SIS test signal. It also includes the automatic functions to isolate the other ventilation systems that are not part of the safety-related postaccident operating configuration and to start up and to align the ventilation systems that flow through the secondary containment to the accident condition.

3/4.6.6.2 SECONDARY CONTAINMENT (continued)

In MODES 5 and 6, the probability and consequences of a DBA are low due to the RCS temperature and pressure limitation in these MODES. Therefore, Secondary Containment is not required in MODES 5 and 6.

ACTIONS

In the event Secondary Containment OPERABILITY is not maintained, Secondary Containment OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a DBA occurring during this time period. Therefore, it is considered that there exists no loss of safety function while in the ACTION Statement.

Inoperability of the Secondary Containment does not make the SLCRS fans and filters inoperable. Therefore, while in this ACTION Statement solely due to inoperability of the Secondary Containment, the conditions and required ACTIONS associated with Specification 3.6.6.1 (i.e., Supplementary Leak Collection and Release System) are not required to be entered. If the Secondary Containment OPERABILITY cannot be restored to OPERABLE status within the required completion time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full-power conditions in an orderly manner and without challenging plant systems.

Surveillance Requirements

4.6.6.2.1

Maintaining Secondary Containment OPERABILITY requires maintaining each door in each access opening in a closed position except when the access opening is being used for normal entry and exit. The normal time allowed for passage of equipment and personnel through each access opening at a time is defined as no more than 5 minutes. The access opening shall not be blocked open. During this time, it is not considered necessary to enter the ACTION statement. A 5-minute time is considered acceptable since the access opening can be quickly closed without special provisions and the probability of occurrence of a DBA concurrent with equipment and/or personnel transit time of 5 minutes is low.

~~The 31-day frequency for this surveillance is based on engineering judgment and is considered adequate in view of the other indications of access opening status that are available to~~ the operator.

Insert 2

AUXILIARY FEEDWATER SYSTEM (Continued)

If all three AFW pumps are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with non safety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW pump to OPERABLE status. Required ACTION e. is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW pump is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

SR 4.7.1.2.1a. verifies the correct alignment for manual, power operated, and automatic valves in the auxiliary feedwater water and steam supply flow paths to provide assurance that the proper flow paths exist for auxiliary feedwater operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulations; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. ~~The 31 day frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~

Insert 2

The SR is modified by a Note that states one or more auxiliary feedwater pumps may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the auxiliary feedwater mode of operation, provided it is not otherwise inoperable. This exception to pump OPERABILITY allows the pump(s) and associated valves to be out of their normal standby alignment and temporarily incapable of automatic initiation without declaring the pump(s) inoperable. Since auxiliary feedwater may be used during STARTUP, SHUTDOWN, HOT STANDBY operations, and HOT SHUTDOWN operations for steam generator level control, and these manual operations are an accepted function of the auxiliary feedwater system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

SURVEILLANCE REQUIREMENTS

For the surveillance requirements, the UHS temperature is measured at the locations described in the LCO write-up provided in this section.

Surveillance Requirement 4.7.5.a verifies that the UHS is capable of providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature. ~~The 24-hour frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.~~ This surveillance requirement verifies that the average water temperature of the UHS is less than or equal to 75°F.

Insert 2

Surveillance Requirement 4.7.5.b requires that the UHS temperature be monitored on an increased frequency whenever the UHS temperature is greater than 70°F during the applicable MODES. The intent of this Surveillance Requirement is to increase the awareness of plant personnel regarding UHS temperature trends above 70°F. ~~The frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.~~

3/4.7.6 DELETED

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

BACKGROUND

The control room emergency ventilation system provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. Additionally, the system provides temperature control for the control room envelope (CRE) during normal and post-accident operations.

The control room emergency ventilation system is comprised of the CRE emergency air filtration system and a temperature control system.

The control room emergency air filtration system consists of two redundant systems that recirculate and filter the air in the CRE and a CRE boundary that limits the inleakage of unfiltered air. Each control room emergency air filtration system consists of a moisture separator, electric heater, prefilter, upstream high efficiency particulate air (HEPA) filter, charcoal adsorber, downstream HEPA filter, and fan. Additionally, ductwork, valves or dampers, and instrumentation form part of the system.

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and other non-critical areas including adjacent support offices,

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

ACTIONS (Continued)

their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

Immediate action(s), in accordance with the LCO ACTION Statements, means that the required action should be pursued without delay and in a controlled manner.

During movement of recently irradiated fuel assemblies

d. With one control room emergency air filtration system inoperable, action must be taken to restore the inoperable system to an OPERABLE status within 7 days. After 7 days, either initiate and maintain operation of the remaining OPERABLE control room emergency air filtration system in the emergency mode or suspend the movement of fuel. Initiating and maintaining operation of the OPERABLE train in the emergency mode ensures: (i) OPERABILITY of the train will not be compromised by a failure of the automatic actuation logic; and (ii) active failures will be readily detected.

e. With both control room emergency air filtration systems inoperable, or with the train required by ACTION 'd' not capable of being powered by an OPERABLE emergency power source, actions must be taken to suspend all operations involving the movement of recently irradiated fuel assemblies. This action places the unit in a condition that minimizes risk. This action does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

4.7.7.a

Insert 2

The CRE environment should be checked periodically to ensure that the CRE temperature control system is functioning properly. ~~Verifying that the CRE air temperature is less than or equal to 95°F at least once per 12 hours is sufficient~~. It is not necessary to cycle the CRE ventilation chillers. The CRE is manned during operations covered by the technical specifications. Typically, temperature aberrations will be readily apparent.

4.7.7.b

Standby systems should be checked periodically to ensure that they function properly. ~~As the environment and normal operating conditions on this system are not too severe, testing the trains~~

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

~~once every 31 days on a STAGGERED TEST BASIS provides an adequate check of this system~~. This surveillance requirement verifies a system flow rate of 1,120 cfm ± 20%. Additionally, the system is required to operate for at least 10 continuous hours with the heaters energized. These operations are sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters due to the humidity in the ambient air.

4.7.7.c as specified in the Surveillance Frequency Control Program

The performance of the control room emergency filtration systems should be checked periodically by verifying the HEPA filter efficiency, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. The frequency is ~~at least once per 24 months~~ or following painting, fire, or chemical release in any ventilation zone communicating with the system.

ANSI N510-1980 will be used as a procedural guide for surveillance testing.

Any time the OPERABILITY of a HEPA filter or charcoal adsorber housing has been affected by repair, maintenance, modification, or replacement activity, post maintenance testing in accordance with SR 4.0.1 is required to demonstrate OPERABILITY.

4.7.7.c.1

This surveillance verifies that the system satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the system at a flow rate of 1,120 cfm ± 20%. ANSI N510-1980 is used in lieu of ANSI N510-1975 referenced in the regulatory guide.

4.7.7.c.2

This surveillance requires that a representative carbon sample be obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978 and that a laboratory analysis verify that the representative carbon sample meets the laboratory testing criteria of ASTM D3803-89 and Millstone Unit 3 specific parameters. The laboratory analysis is required to be performed within 31 days after removal of the sample. ANSI N510-1980 is used in lieu of ANSI N510-1975 referenced in Revision 2 of Regulatory Guide 1.52.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.c.3

This surveillance verifies that a system flow rate of 1,120 cfm ± 20%, during system operation when testing in accordance with ANSI N510-1980.

4.7.7.d

After 720 hours of charcoal adsorber operation, a representative carbon sample must be obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and a laboratory analysis must verify that the representative carbon sample meets the laboratory testing criteria of ASTM D3803-89 and Millstone Unit 3 specific parameters.

The laboratory analysis is required to be performed within 31 days after removal of the sample. ANSI N510-1980 is used in lieu of ANSI N510-1975 referenced in Revision 2 of Regulatory Guide 1.52.

The maximum surveillance interval is 900 hours, per Surveillance Requirement 4.0.2. The 720 hours of operation requirement originates from Nuclear Regulatory Guide 1.52, Table 2, Note C. This testing ensures that the charcoal adsorbency capacity has not degraded below acceptable limits as well as providing trending data.

4.7.7.e.1

This surveillance verifies that the pressure drop across the combined HEPA filters and charcoal adsorbers banks at less than 6.75 inches water gauge when the system is operated at a flow rate of 1,120 cfm ± 20%. ~~The frequency is at least once per 24 months~~.

4.7.7.e.2

Deleted.

Insert 2

4.7.7.e.3

This surveillance verifies that the heaters can dissipate 9.4 ± 1 kW at 480V when tested in accordance with ANSI N510-1980. ~~The frequency is at least once per 24 months~~. The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage.

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM

Insert 2

Periodic

The OPERABILITY of the Auxiliary Building Filter System, and associated filters and fans, ensures that radioactive materials leaking from the equipment within the charging pump, component cooling water pump and heat exchanger areas following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating for at least 10 continuous hours ~~in a 31 day period~~ is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Laboratory testing of methyl iodide penetration shall be performed in accordance with ASTM D3803-89 and Millstone Unit 3 specific parameters. The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage.

The Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation System is required to be available to support the Auxiliary Building Filter System and the Supplementary Leak Collection and Release System (SLCRS). The Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation System consists of two redundant trains, each capable of providing 100% of the required flow. Each train has a two position, "Off" and "Auto," remote control switch. With the remote control switches for each train in the "Auto" position, the system is capable of automatically transferring operation to the redundant train in the event of a low flow condition in the operating train. The associated fans do not receive any safety related automatic start signals (e.g. Safety Injection Signal).

Placing the remote control switch for a Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation Train in the "Off" position to start the redundant train or to perform post maintenance testing to verify availability of the redundant train will not affect the availability of that train, provided appropriate administrative controls have been established to ensure the remote control switch is immediately returned to the "Auto" position after the completion of the specified activities or in response to plant conditions. These administrative controls include the use of an approved procedure and a designated individual at the control switch for the respective Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation Train who can rapidly respond to instructions from procedures, or control room personnel, based on plant conditions.

3/4.8 ELECTRICAL POWER SYSTEMS

BASES

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION

Technical Specification 3.8.1.1.b.1 requires each of the diesel generator day tanks contain a minimum volume of 278 gallons. Technical Specification 3.8.1.2.b.1 requires a minimum volume of 278 gallons be contained in the required diesel generator day tank. This capacity ensures that a minimum usable volume of 189 gallons is available. This volume permits operation of the diesel generators for approximately 27 minutes with the diesel generators loaded to the 2,000 hour rating of 5335 kw. Each diesel generator has two independent fuel oil transfer pumps. The shutoff level of each fuel oil transfer pump provides for approximately 60 minutes of diesel generator operation at the 2000 hour rating. The pumps start at day tank levels to ensure the minimum level is maintained. The loss of the two redundant pumps would cause day tank level to drop below the minimum value.

Technical Specification 3.8.1.1.b.2 requires a minimum volume of 32,760 gallons be contained in each of the diesel generator's fuel storage systems. Technical Specification 3.8.1.2.b.2 requires a minimum volume of 32,760 gallons be contained in the required diesel generator's fuel storage system. This capacity ensures that a minimum usable volume (29,180 gallons) is available to permit operation of each of the diesel generators for approximately three days with the diesel generators loaded to the 2,000 hour rating of 5335 kW. The ability to cross-tie the diesel generator fuel oil supply tanks ensures that one diesel generator may operate up to approximately six days. Additional fuel oil can be supplied to the site within twenty-four hours after contacting a fuel oil supplier.

Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power source and distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Surveillance Requirements 4.8.1.1.2.a.6 ~~(monthly) and~~ 4.8.1.1.2.b.2 ~~(once per 184 days)~~ and 4.8.1.1.2.j ~~(18 months test)~~

The Surveillances 4.8.1.1.2.a.6 and 4.8.1.1.2.b.2 verify that the diesel generators are capable of synchronizing with the offsite electrical system and loaded to greater than or equal to continuous rating of the machine. A minimum time of 60 minutes is required to stabilize engine temperatures, while

minimizing the time that the diesel generator is connected to the offsite source. Surveillance Requirement 4.8.1.1.2.j requires demonstration ~~once per 18 months~~ that the diesel generator can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ 2 hours of which are at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the diesel generator. The load band is provided to avoid routine overloading of the diesel generator. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain diesel generator OPERABILITY. The load band specified accounts for instrumentation inaccuracies, operational control capabilities, and human factor characteristics. The note (*) acknowledges that a momentary transient outside the load range shall not invalidate the test.

Surveillance Requirements 4.8.1.1.2.a.5 ~~(Monthly)~~, 4.8.1.1.2.b.1 ~~(Once per 184 Days)~~, 4.8.1.1.2.g.4.b ~~(18 Month Test)~~, 4.8.1.1.2.g.5 ~~(18 Month Test)~~ and 4.8.1.1.2.g.6.b ~~(18 Month Test)~~

Several diesel generator surveillance requirements specify that the emergency diesel generators are started from a standby condition. Standby conditions for a diesel generator means the diesel engine coolant and lubricating oil are being circulated and temperatures are maintained within design ranges. Design ranges for standby temperatures are greater than or equal to the low temperature alarm setpoints and less than or equal to the standby "keep-warm" heater shutoff temperatures for each respective sub-system.

Insert 2

Surveillance Requirement 4.8.1.1.2.j ~~(18 Month Test)~~

The existing "standby condition" stipulation contained in specification 4.8.1.1.2.a.5 is superseded when performing the hot restart demonstration required by 4.8.1.1.2.j.

Any time the OPERABILITY of a diesel generator has been affected by repair, maintenance, or replacement activity, or by modification that could affect its interdependency, post maintenance testing in accordance with SR 4.0.1 is required to demonstrate OPERABILITY.

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1975 & 1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations." Sections 5 and 6 of IEEE Std 450-1980 replaced Sections 4 and 5 of IEEE Std 450-1975. Guidance on bypassing weak cells, if required, is in accordance with section 7.4 of IEEE 450-2002. The balance of IEEE Std 450-1975 applies. Insert 2

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8-2a specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2a is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

If the required power sources or distribution systems are not OPERABLE in MODES 5 and 6, operations involving CORE ALTERATIONS, positive reactivity changes, movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the

3/4.9.8.1 HIGH WATER LEVEL (continued)

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operations, except as permitted in the Note to the LCO.

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

Surveillance Requirement

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. ~~The frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR system.~~ Insert 2

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

Surveillance Requirement

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. ~~The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.~~

Insert 2