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10 CFR 50.90

NMP1L3027
May 12, 2015

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

Nine Mile Point Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-63
NRC Docket No. 50-220

SUBJECT: Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

On March 26, 2015, Exelon Generation Company, LLC (Exelon) submitted an amendment to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. DPR-63 for Nine Mile Point Nuclear Station Unit 1 (NMP-1). Exelon has determined that the March 26, 2015, letter requires minor editorial clarifications to the cover letter and the attachment containing the proposed TS and Bases page changes. This submittal provides these clarifications, and supersedes in its entirety the previously submitted request dated March 26, 2015.

In accordance with the provisions of Title 10 of the Code of Federal Regulations (10 CFR 50.90), "Application for amendment of license, construction permit, or early site permit," Exelon is submitting a request for an amendment to the Technical Specifications (TS), Appendix A, of Renewed Facility Operating License No. DPR-63 for Nine Mile Point Nuclear Station Unit 1 (NMP-1).

The proposed amendment would modify NMP-1 TS by relocating specific surveillance frequencies to a licensee-controlled program with the implementation of Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, and Risk-Informed Method for Control of Surveillance Frequencies."

The changes are consistent with NRC-approved Industry Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b, 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.

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Attachment 1 provides a description of the proposed change, the requested confirmation of applicability, and plant-specific verifications. Attachment 2 provides documentation of Probabilistic Risk Assessment (PRA) technical adequacy. Attachments 3 provides the existing NMP Unit 1, TS and TS Bases pages marked up to show the proposed changes. Attachment 4 provides a TSTF-425 versus NMP Unit 1 TS Cross-Reference. Attachment 5 provides the proposed No Significant Hazards Consideration. Attachment 6 provides the proposed Inserts.

There are no regulatory commitments contained in this letter.

Exelon requests approval of the proposed license amendment by May 12, 2016, with the amendment being implemented within 120 days.

These proposed changes have been reviewed by the Plant Operations Review Committee and approved in accordance with Nuclear Safety Review Board procedures.

In accordance with 10 CFR 50.91, "Notice for Public Comment; State Consultation," a copy of this application, with attachments, is being provided to the designated State Official.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 12th day of May 2015.

If you should have any questions regarding this submittal, please contact Enrique Villar at 610-765-5736.

Respectfully,

for

James Barstow
Director - Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
1. Description and Assessment
2. Documentation of PRA Technical Adequacy
3. Proposed Technical Specification and Bases Page Changes
4. TSTF-425 (NUREG-1433) vs. NMP-1 Cross-Reference
5. Proposed No Significant Hazards Consideration
6. Proposed Inserts

cc:	USNRC Regional Administrator, Region I	w/attachments
	USNRC Senior Resident Inspector - NMP	"
	USNRC Project Manager, NRR - NMP	"
	A. L. Peterson, NYSERDA	"

ATTACHMENT 1

License Amendment Request

Nine Mile Point Nuclear Station, Unit 1
Docket No. 50-220

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program
(Adoption of TSTF-425, Revision 3)

Description and Assessment

DESCRIPTION AND ASSESSMENT

1.0 DESCRIPTION

The proposed amendment would modify the Nine Mile Point Nuclear Station Unit 1 (NMP-1) Technical Specifications (TS) 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" (Ref. 1). Additionally, the change would add a new program, the Surveillance Frequency Control Program, 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) (Ref. 2), announced the availability of this TS improvement.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Exelon Generation Company, LLC (Exelon) has reviewed the NRC staff's Model Safety Evaluation for TSTF-425, Revision 3, dated July 6, 2009. This review included a review of the NRC staff's Model Safety Evaluation, TSTF-425, Revision 3, and the requirements specified in NEI 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," (ADAMS Accession No. ML071360456) (Ref. 3).

The traveler and Model Safety Evaluation discuss the applicable regulatory requirements and guidance, including the 10 CFR 50, Appendix A, General Design Criteria (GDC). NMP-1 was not licensed to the 10 CFR 50, Appendix A GDC. The NMP-1 Updated Final Safety Analysis Report (UFSAR) provides an assessment against the GDC in Table I-1. This UFSAR table is a summary and refers to the NMP-1 Technical Supplement to Petition for Conversion from Provisional Operating License to Full-Term Operating License, July 1972. The Technical Supplement provides an analysis of plant design criteria for NMP-1 to the GDC criteria. Based on the analysis performed in the Technical Supplement, Exelon believes that the plant-specific requirements for NMP-1 are sufficiently similar to the Appendix A GDC and represent an adequate technical basis for adopting the proposed change.

Attachment 2 includes Exelon's documentation with regard to Probabilistic Risk Assessment (PRA) technical adequacy consistent with the requirements of Regulatory Guide 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (ADAMS Accession No. ML070240001) (Ref. 4), Section 4.2, and describes any PRA models without NRC-endorsed standards, including documentation of the quality characteristics of those models in accordance with Regulatory Guide 1.200.

Exelon has concluded that the justifications presented in the TSTF proposal and the NRC staff's Model Safety Evaluation prepared by the NRC staff are applicable to NMP-1 and justify this amendment to incorporate the changes to the NMP-1 TS.

2.2 Optional Changes and Variations

The proposed amendment is consistent with the STS changes described in TSTF-425, Revision 3; however, Exelon proposes variations or deviations from TSTF-425, as identified below, which includes differing Surveillance numbers.

Revised (clean) 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 NMP-1 amendment requests that will impact some of the same TS pages. Providing only mark-ups of the proposed TS changes satisfies the requirements of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," (Ref. 5) in that the mark-ups fully describe the changes desired. This is 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. Also, since the Bases for the NMP-1 Surveillance Requirements are intermingled throughout the TS Surveillance Requirement sections, mark-ups of both the proposed TS changes and the proposed TS Bases changes are provided together in Attachment 3.

Attachment 4 provides a cross-reference between the NUREG-1433 Surveillances included in TSTF-425 versus the NMP-1 TS Surveillances included in this amendment request. Attachment 4 includes a summary description of the referenced TSTF-425 (NUREG-1433)/NMP-1 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-1433 TS Surveillances included in TSTF-425 and corresponding NMP-1 Surveillances have differing Surveillances numbers,
b. NUREG-1433 TS Surveillances included in TSTF-425 that are not contained in the NMP-1 TS, and
c. NMP-1 plant-specific TS Surveillances that are not contained in NUREG-1433 and, therefore, are not included in the TSTF-425 mark-ups.

Concerning the above, NMP-1 TS are custom TS for a Boiling Water Reactor (BWR) plant. As a result, the applicable NMP-1 TS and associated Bases wording, format and numbering differ from the Standard Technical Specifications (STS) presented in NUREG-1433 and TSTF-425, Revision 3. However, the applicable NMP-1 TS Surveillances and associated Bases are equivalent to the NUREG-1433 and TSTF-425, Revision 3, in that they assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Therefore, this deviation 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).

In addition, there are Surveillances contained in NUREG-1433 that are not contained in the NMP-1 TS. Therefore, the NUREG-1433 mark-ups included in TSTF-425 for these TS Surveillances are not applicable to NMP-1. 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).

Furthermore, the NMP-1TS include plant-specific TS Surveillances that are not contained in NUREG-1433 and, therefore, are not included in the NUREG-1433 mark-ups provided in TSTF-425. Exelon has determined that the relocation of the frequencies for these NMP-1 plant-specific TS Surveillances is consistent with TSTF-425, Revision 3, and with the NRC staff'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. Changes to the frequencies for these plant-specific TS Surveillances would be controlled under the Surveillance Frequency Control Program (SFCP). The SFCP provides the necessary administrative controls to require that TS Surveillances related to testing, calibration, and inspection are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the Limiting Conditions for Operation will be met. Changes to frequencies in the SFCP would be evaluated using the methodology and probabilistic risk 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 systems, components, and structures (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 Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998 (ADAMS Accession No. ML003740176) (Ref. 6), relative to changes in Surveillance Frequencies. Therefore, the proposed relocation of the NMP-1 plant-specific TS Surveillance Frequencies is consistent with TSTF-425 and with the NRC staff's Model Safety Evaluation, dated July 6, 2009 (74 FR 31996).

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration

Exelon has reviewed the proposed No Significant Hazards Consideration (NSHC) determination published in the Federal Register dated July 6, 2009 (74 FR 31996). Exelon has concluded that the proposed NSHC presented in the Federal Register Notice is applicable to NMP-1, and is provided as Attachment 4 to this amendment request, which satisfies the requirements of 10 CFR 50.91(a), "Notice for public comment; State consultation" (Ref. 7).

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 (ADAMS Accession No. ML090850642) and the NRC staff's Model Safety Evaluation published in the Notice of Availability, dated July 6, 2009 (74 FR 31996). Exelon has concluded that the relationship of the proposed changes to the applicable regulatory requirements presented in the Federal Register Notice is applicable to NMP-1.

3.3 Precedence

This application is being made in accordance with the TSTF-425, Revision 3 (ADAMS Accession No. ML090850642). Exelon is not proposing significant variations or deviations from the TS changes described in TSTF 425 or in the content of the NRC staff's Model Safety Evaluation published on July 6, 2009 (74 FR 31996). The NRC has previously approved amendments to the TS as part of the pilot process for TSTF-425, including Amendment No. 276 for Oyster Creek Nuclear Power Station, dated September 27, 2010. Similar to the NMP – 1 TS, the Oyster Creek license amendment involved custom TS.

3.4 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

Exelon 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). Exelon has concluded that the staff's findings presented therein are applicable to NMP-1, 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. 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit."

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

7. 10 CFR 50.91(a), "Notice for public comment; State consultation."

ATTACHMENT 2

License Amendment Request

Nine Mile Point Nuclear Station, Unit 1
Docket No. 50-220

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

Documentation of PRA Technical Adequacy

Documentation of PRA Technical Adequacy

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Documentation of PRA Technical Adequacy

2.1 Overview

The implementation of the Surveillance Frequency Control Program (also referred to as Technical Specifications Initiative 5b) at Nine Mile Point Unit 1 (NMP-1) will follow the guidance provided in NEI 04-10, Revision 1 [Ref. 1] in evaluating proposed TS Surveillance Test Interval (STI; also referred to as "surveillance frequency") changes.

The following steps of the risk-informed STI revision process are common to proposed changes to all STIs within the proposed licensee-controlled program.

- Each STI revision is reviewed to determine whether there are any commitments made to the NRC that may prohibit changing the interval. If there are no related commitments, or the commitments may be changed using a commitment change process based on NRC endorsed guidance, then evaluation of the STI revision would proceed. If a commitment exists and the commitment change process does not permit the change, then the STI revision would not be implemented.
- 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 testing and system or component reliability. If the IDP approves the STI revision, the change is documented and implemented, and available for audit by the Nuclear Regulatory Commission (NRC). If the IDP does not approve the STI revision, the surveillance frequency is left unchanged.
- Performance monitoring is conducted as recommended by the IDP. In some cases, no additional monitoring may be necessary beyond that already conducted under the Maintenance Rule. The performance monitoring helps to confirm that no failure mechanisms related to the revised STI become important enough to alter the information provided for the justification of the interval changes.
- The IDP is responsible for periodic review of performance monitoring results. If it is determined that the time interval between successive performances of a surveillance test is a factor in the unsatisfactory performances of the surveillance, the IDP returns the STI 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. Also, the cumulative impact of all risk-informed STI revisions on all PRAs (i.e., internal events, external events and shutdown) is also compared to the risk acceptance criteria as delineated in NEI 04-10.

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 NEI 04-10 [Ref. 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 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
   - 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 and external), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the 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, in RG-1.200 Revision 1 this is just 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.

2.2 Technical Adequacy of the PRA Model

The NM114B version of the NMP-1 PRA model is the most recent evaluation of internal event risks. The NMP-1 PRA modeling 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 NMP-1 PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

Exelon Generation Company, LLC (Exelon) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Exelon nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. Prior to joining the Exelon nuclear fleet in 2014, comparable practices were in place when NMP was owned and operated by Constellation Energy Nuclear Group (CENG). Because of the similarities between the CENG and Exelon practices, no additional discussion specifically regarding the legacy CENG approach will be provided. The following information describes the Exelon approach (and by extension the CENG approach) to PRA model maintenance, as it applies to the NMP-1 PRA.

PRA Maintenance and Update

The Exelon risk management process ensures that the applicable PRA model is an accurate reflection of the as-built and as-operated plants. This process is defined in the Exelon Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation Training and Reference Materials (T&RM's).

- Exelon procedure ER-AA-600-1015, "Full Power Internal Event (FPIE) PRA Model Update," delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Exelon nuclear generation sites.
- ER-AA-600-1061 "Fire PRA Model Update and Control" delineates the responsibilities and guidelines for updating the station fire PRA.

The overall Exelon Risk Management program, including ER-AA-600-1015 and ER-AA-600-1061, define the process for: implementing regularly scheduled and interim PRA model updates; for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, industry operating experience, etc.); and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated during each model update.

In addition to these activities, Exelon risk management procedures provide the guidance for particular risk management maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for Exelon nuclear generation sites.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on Systems, Structures, and Components (SSCs) within the scope of the Maintenance Rule (10 CFR 50.65(a)(4)).

An application specific update of the PRA model was completed in the 4th quarter of 2014 to support application of the NFPA-805 process. Exelon will be performing a regularly scheduled FPIE model update to the NMP-1 PRA in 2015, which is expected to be approved in the first half of 2016.

As indicated previously, RG 1.200 also requires that additional information be provided as part of the License Amendment Request (LAR) submittal to demonstrate the technical adequacy of the PRA model used for the risk assessment. Each of these items (plant changes not yet incorporated in to the PRA model, relevant peer review findings, consistency with applicable PRA Standards, and the identification of key assumptions) will be discussed in turn.

2.2.1 Plant Changes Not Yet Incorporated into the PRA Model

Each Exelon station maintains an Updating Requirements Evaluation (URE) database to track all enhancements, corrections, and unincorporated plant changes. During the normal screening conducted as part of the plant change process, if a potential model update is identified a new URE database item is created. Depending on the potential impact of the identified change, the requirements for incorporation will vary.

As part of the PRA evaluation for each STI change request, a review of open items in the URE database for NMP-1 will be performed and an assessment of the impact on the results of the application will be made prior to presenting the results of the risk analysis to the IDP. If the STI represents a potential impact to the PRA, then performance of additional sensitivity studies or PRA model changes to confirm the impact on the risk analysis may be considered.

2.2.2 Applicability of Peer Review Findings and Observations

A PRA model update was completed in 2008, resulting in the NM114A updated model. The NMP1 PRA model was revised to meet RG 1.200, revision 1, guidance and comply with the ASME/ANS PRA Standard RA-Sc-2007 [Ref. 4]. This revision included a significant change in methodology, converting from RISKMAN (Large Event Trees, Small Fault Trees) to CAFTA (Small Event Trees, Large Fault Trees).

This model/methodology change was peer reviewed under the auspices of the BWR Owners Group (BWROG) in the 1st quarter of 2008 [Ref. 9]. This peer review was performed following the Industry PRA Peer Review process [Ref. 3], NEI 05-04 [Ref. 7], and NEI 00-002 [Ref. 8]. This peer review included an assessment of the PRA model maintenance and update process.

Since the 2008 peer review, an application specific PRA model update was completed in 2014 to support implementation of NFPA-805. As part of the development of this model a peer review of the fire PRA was conducted in January of 2012 [Ref. 10]. This peer review used NEI 07-12 [Ref. 11] to evaluate the model against the ASME PRA Standard (ASME/ANS RA-Sa-2009) [Ref. 12] along with the NRC clarifications provided in Regulatory Guide 1.200, Rev. 2.The new model is the NM114B updated model.

2.2.3 Consistency with Applicable PRA Standards

As indicated above there have been two relevant peer reviews conducted on the current PRA model.

- The 2008 peer review for the PRA ASME model update identified 309 Supporting Requirements (SR) applicable to the NMP-1 PRA. Of these: 29 were not met; 2 met capability category (cc) 1; 13 partially met cc 2; 31 met cc 2; 22 partially met cc 3; 14 met cc3; and, 198 fully met all capability requirements. There were 26 findings and observations (F&O's) issued to address the identified gaps to compliance with the PRA standard. Subsequent to the peer review, 21 of the findings have been addressed, 5 findings are still open pending the next model update. Those F&O's still open are listed in Table 2-1.
- The 2012 fire PRA peer review for the PRA ASME model update identified 279 SRs applicable to the NMP-1 PRA. Of these: 37 were not met; 5 met capability category (cc) 1; 10 partially met cc 2; 29 met cc 2; 16 partially met cc 3; 5 met cc 3; and, 177 fully met all capability requirements. There were 65 findings and 59 suggestions issued to address potential gaps to compliance with the PRA standard. Subsequent to the peer review, in order to satisfy the NFPA-805 transition LAR, all of the findings and suggestions have been addressed. As the results of this peer review have already been communicated to the NRC as part of the NFPA-805 submittal [Ref. 13] [Ref. 14] [Ref. 15] and subsequent requests for additional information (RAI) [Ref. 16] [Ref. 17] [Ref. 18] [Ref. 19] [Ref. 20] [Ref. 21] [Ref. 22] [Ref. 23] [Ref. 24] [Ref. 25] [Ref. 26]; these will not be catalogued in this document.

All remaining gaps will be reviewed for consideration during the 2015 model update but are judged to have low impact on the PRA model or its ability to support a full range of PRA applications. The remaining gaps are documented in the URE database so that they can be tracked and their potential impacts accounted for in applications where appropriate.

Each item will be reviewed as part of each STI change assessment that is performed and an assessment of the impact on the results of the application will be made prior to presenting the results of the risk analysis to the IDP. If the STI represents a potential impact to the PRA, then performance of additional sensitivity studies or PRA model changes to confirm the impact on the risk analysis may be considered.

Table 2-1: Open Gaps from ASME PRA Standard Peer Review and Impact on T.S. Frequency Control Program (TSTF-425)

URE #	DESCRIPTION OF GAP	SR	IMPORTANCE TO APPLICATION
502	The IE notebook (as well as the other PRA notebooks) contains a discussion of the assumptions used. The notebooks also provide a discussion of plant-specific sources of uncertainty. However, this SR requires a systematic evaluation of all sources of uncertainty, including industrywide issues with data and modeling approaches. Recent EPRI reports are available that document generic industry uncertainty sources. These items should be reviewed for applicability to NMP1 and added to the PRA documentation.	IE-D03	This gap will is necessarily addressed in each STI evaluation per steps 5 and 14 of NEI-04-10. Since an explicit evaluation is required for each submittal, this finding does not adversely impact the capability to implement TSTF-425.
515	Appendices to the QU notebook present the top 200 CDF and LERF cutsets. The top 20 CDF cutsets are specifically discussed in section 4.2.3 in the context of plant response, significant assumptions made, etc. While the top 200 cutsets are included, the analysis of these cutsets should be expanded to include a greater number of cutsets, as the top 20 only constitutes about 60% of the CDF and is dominated by cutsets with only an initiator and one failure (i.e., does not demonstrate a comprehensive review)	QU-D1a	As part of each STI evaluation, sensitivity analyses will be required to validate adequacy of the risk evaluation performed. As part of these evaluations, the cutsets generated will be reviewed. This finding will not interfere with the capability to perform this review. This finding does not adversely impact the capability to implement TSTF-425.
519	Some insights from the importance listings (for equipment and operator actions) are discussed in section 4.2.4 of QU NB However, further discussion should be provided to specifically address the requirements of this SR. Provide a more detailed discussion how this SR is met. QU NB does not have adequate discussion.	QU-D5b	This finding will not interfere with the capability to perform this review. This finding does not adversely impact the capability to implement TSTF-425.
522	Section 4.3.5 of the QU notebook briefly notes that a review was performed. However, there is no evidence presented in the notebook. The QU notebook should include a sampling of several non-significant cutsets and demonstrate that these cutsets correctly	QU-D04	This finding will not interfere with the capability to perform this review. This finding does not adversely impact the capability to implement TSTF-425.
523	Section 5.2 in QU NB addresses this issue qualitatively. More sensitivity runs need to be done to evaluate model uncertainties, e.g., Set all HEPs, CCFs etc at 5th and 95th percentile during quantification.	QU-E04	This finding will not interfere with the capability to perform this review. This finding does not adversely impact the capability to implement TSTF-425.

2.2.4 Identification of Key Assumptions

The overall Initiative 5b process is a risk-informed process with the PRA model results providing one of the inputs to the IDP to determine if an STI change is warranted. The methodology recognizes that a key area of uncertainty for this application is the standby failure rate utilized in the determination of the STI extension impact. Therefore, the methodology requires the performance of selected sensitivity studies on the standby failure rate of the component(s) of interest for the STI assessment.

The results of the standby failure rate sensitivity study plus the results of any additional sensitivity studies identified during the performance of the reviews as outlined in 2.2.1 and 2.2.3 above (including a review of identified sources of uncertainty that were developed for NMPU1 based on the EPRI 1016737 guidance [Ref. 27]) for each STI change assessment will be documented and included in the results of the risk analysis that goes to the IDP.

2.3 External Events Considerations

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

External hazards were evaluated in the NMPU1 Individual Plant Examination of External Events (IPEEE) submittal in response to the NRC IPEEE Program (Generic Letter 88-20, Supplement 4) [Ref. 6]. The IPEEE Program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of associated severe accident risks.

The results of the NMPU1 IPEEE study are documented in the NMPU1 IPEEE Main Report [Ref. 28]. The primary areas of external event evaluation at NMPU1 were internal fire and seismic initiating events. The internal fire events were addressed by using a modified version of the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology [Ref. 29] and the seismic evaluations were performed in accordance with the EPRI Seismic Margins Analysis (SMA) methodology [Ref. 30]. As such, there are no comprehensive CDF and LERF values available from the IPEEE to support the STI risk assessments.

In addition to internal fires and seismic events, the NMPU1 IPEEE analysis of high winds, floods, and other (HFO) external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Since NMPU1 was designed (with construction started) prior to the issuance of the 1975 Standard Review Plan (SRP) criteria, Niagara Mohawk [now Exelon] performed a plant hazard and design information review for conformance with the SRP criteria. For seismic and fire events that were not screened out, additional analyses were performed to determine whether or not the hazard frequency was acceptably low. HFO events were screened out by compliance with the 1975 SRP criteria. As such, these hazards were determined in the NMPU1 IPEEE to be negligible contributors to overall plant risk.

Since the performance of the IPEEE, the NRC approved conversion from appendix R compliance to NFPA-805 for fire protection [Ref. 14]. Pursuant to this change, a fire PRA has been created and implemented at NMPU1. This Fire PRA model was created under the auspices of NUREG/CR-6850 [Ref. 31] and has undergone BWROG peer review (completed February 2012) [Ref. 13]. The National Institute of Standards and Technology (NIST)

Consolidated Model of Fire and Smoke Transport (CFAST) Methodology [Ref. 33], EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [Ref. 31], Fire Events Database [Ref. 34] and plant specific data were used to develop the NMP1 Fire PRA. This fire PRA has numerous capabilities not considered in the IPEEE fire PRA model, including explicit analysis of the main control room (MCR) and cable spreading room (CSR), as well as multiple spurious operations (MSO) considerations. The ignition frequencies for the MCR and CSR were developed using the guidance in NUREG/CR-6850 [Ref. 31]. This Fire PRA model has been integrated with the internal events model to create NM114B. Ignition frequencies will be based on the NUREG/CR-6850 [Ref. 31] methodologies and incorporate revised guidance and ignition frequencies [Ref. 32].

As stated earlier, the NEI 04-10 [Ref. 1] methodology allows for STI change evaluations to be performed in the absence of quantifiable PRA models for all external hazards. Therefore, in performing the assessments for the other hazard groups, a qualitative or a bounding approach will be utilized in most cases. Where applicable, the results of any STI change will be evaluated against this model to ensure there is no undue risk associated with a given STI change. This approach is consistent with the accepted NEI 04-10 methodology.

2.4 Summary

The NMPU1 PRA technical capability evaluations and the maintenance and update processes described above provide a robust basis for concluding that the NMPU1 PRA is suitable for use in risk-informed processes such as that proposed for the implementation of a Surveillance Frequency Control Program. Also, in addition to the standard set of sensitivity studies required per the NEI 04-10 [Ref. 1] methodology, open items for changes at the site and remaining gaps to specific requirements in the PRA standard will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

2.5 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] Boiling Water Reactors Owners' Group, BWROG PSA Peer Review Certification Implementation Guidelines, Revision 3, January 1997.

[4] American Society of Mechanical Engineers, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME RA-Sc-2007, New York, New York, July 2007.

[5] U.S. Nuclear Regulatory Commission, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Draft Regulatory Guide DG-1122, November 2002.

[6] NRC Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4," June 28, 1991.

[7] NEI 05-04, Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard.

[8] NEI 00-002 Probabilistic Risk Assessment (PRA) Peer Review Process Guidance, Revision 1, Nuclear Energy Institute (NEI), Washington, DC, May 2006.

[9] Nine Mile Point Unit 1 Plant 2008 RG1.200 PRA Peer Review. May 2008.

[10] Nine Mile Point Unit 1 Plant 2012 RG1.200 PRA Peer Review. January 2012.

[11] NEI 07-12, Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines, Revision 1, Nuclear Energy Institute (NEI), Washington, DC, June 2010.

[12] American Society of Mechanical Engineers, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME PRA Standard (ASME/ANS RA-Sa-2009), New York, New York, July 2009.

[13] U.S. Nuclear Regulatory Commission, "Record of Review, Nine Mile Point Nuclear Station, Unit 1, LAR Attachment U - Table U-1 Internal Events PRA Peer Review - Facts and Observations (F&Os)," and "Record of Review, Nine Mile Point Nuclear Station, Unit 1, LAR Attachment V - Tables V-1 and V-2 Fire PRA Peer Review - Facts and Observations (F&Os)," April, 2014 and March 12, 2014 (ADAMS Accession Nos. ML14122A253 and ML14122A254, respectively).

[14] Bhalchandra Vaidya, U.S. Nuclear Regulatory Commission, letter to Mr. Christopher Costanzo (NMPNS), dated June 30, 2014, NINE MILE POINT NUCLEAR STATION, UNIT NO.1 - ISSUANCE OF AMENDMENT REGARDING TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION PROGRAM IN ACCORDANCE WITH 10 CFR 50.48(c) (TAG NO. ME8899).

[15] Letter from K. Langdon (NMPNS) to Document Control Desk (NRC), dated June 11, 2012, License Amendment Request Pursuant to 10 CFR 50.90: Adoption of NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants {2001 Edition) (ADAMS Accession Nos. ML12170A868 and ML12170A869).

[16] Letter from Christopher Costanzo (NMPNS) to Document Control Desk (NRC), dated February 27, 2013, License Amendment Request Pursuant to 10 CFR 50.90: Adoption of NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) - Response to NRC Request for Additional Information (ADAMS Accession No. ML13064A466}.

[17] Letter from Christopher Costanzo (NMPNS) to Document Control Desk (NRC), dated March 27, 2013, License Amendment Request Pursuant to 10 CFR 50.90: Adoption of NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) - Response to NRC Request for Additional Information {TAC No. ME8899} (ADAMS Accession No. ML13092A139).

[18] Letter from Christopher Costanzo (NMPNS) to Document Control Desk (NRC), dated April 30, 2013 License Amendment Request Pursuant to 10 CFR 50.90: Adoption of NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) - Response to NRC Request for Additional Information (ADAMS Accession Nos. ML131270405 (package), and ML13127A395, ML13127A397, ML13127A398) Portions contain proprietary SUNS/, withheld under 10 CFR 2. 390.

[19] Letter from Christopher Costanzo (NMPNS) to Document Control Desk (NRC), dated December 9, 2013 License Amendment Request Pursuant to 10 CFR 50.90: Adoption of NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) - Response to NRC Request for Additional Information (ADAMS Accession No. ML13347B187).

[20] Letter from Terry Syrell (NMPNS) to Document Control Desk (NRC), dated January 22, 2014, License Amendment Request Pursuant to 10 CFR 50.90: Adoption of NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) - Response to NRC Request for Additional Information (ADAMS Accession No ML14024A416).

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[22] Letter from Christopher Costanzo (NMPNS) to Document Control Desk (NRC), dated April 15, 2014, License Amendment Request Pursuant to 10 CFR 50.90: Adoption of NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) - Response to NRC Request for Additional Information (ADAMS Accession No. ML14113A175).

[23] Letter from Christopher Costanzo (NMPNS) to Document Control Desk (NRC), dated May 09, 2014, License Amendment Request Pursuant to 1 0 CFR 50.90: Adoption of NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) - Response to NRC Request for Additional Information (ADAMS Accession No ML14132A199).

[24] Bhalchandra Vaidya, U.S. Nuclear Regulatory Commission, letter to Mr. Ken Langdon (NMPNS), dated January 3, 2013, Nine Mile Point Nuclear Station, Unit No.1 - Request For Additional Information Regarding License Amendment Request For Adoption of NFPA 805 (ADAMS Accession No. ML12361A050).

[25] Bhalchandra Vaidya, U.S. Nuclear Regulatory Commission, letter to Christopher Costanzo (NMPNS) dated October 9, 2013, Nine Mile Point Nuclear Station, Unit No. 1 - Second Round Of Request For Additional Information Regarding License Amendment Request for Adoption of NFPA 805 (ADAMS Accession No. ML13281A010).

[26] Bhalchandra Vaidya, U.S. Nuclear Regulatory Commission, letter to Christopher Costanzo (NMPNS) dated February 12, 2014, Nine Mile Point Nuclear Station, Unit No. 1 -Third Round Of Request For Additional Information Regarding License Amendment Request for Adoption of NFPA 805 (ADAMS Accession No. ML14030A377).

[27] Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, EPRI, Palo Alto, CA: December 2008 (Final). 1016737.
[28] NINE MILE POINT NUCLEAR STATION - UNIT 1 INDIVIDUAL PLANT EXAMINATION for EXTERNAL EVENTS (IPEEE), Main Report, August 1996.
[29] Professional Loss Control, Inc., Fire-Induced Vulnerability Evaluation (FIVE) Methodology Plant Screening Guide, EPRI TR-100370, Electric Power Research Institute, Final Report, April 1992.
[30] NTS Engineering, et. al., A Method for Assessment of Nuclear Power Plant Seismic Margin, EPRI NP-6041, Electric Power Research Institute, Final Report, August 1991.
[31] EPRI/NRC-RES, Fire PRA Methodology for Nuclear Power Facilities, EPRI 1011989, NUREG/CR-6850, Final Report, September 2005.
[32] Fire Probabilistic Risk Assessment Methods Enhancements Supplement 1 to NUREG/CR-6850 and EPRI 1011989, EPRI 1019259, Electric Power Research Institute, December 2009.
[33] National Institute of Standards and Technology's (NIST) Consolidated Model of Fire Growth and Smoke Transport (CFAST) Version (6) (Jones et al., 2004).
[34] NSAC/179L, Electric Power Research Institute, Fire Events Database for U.S. Nuclear Power Plants, Rev. 1, January, 1993.

ATTACHMENT 3

License Amendment Request

Nine Mile Point Nuclear Station, Unit 1
Docket No. 50-220

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

Proposed Technical Specification and Bases Page Changes

30	78	159	197	244
33	96	160	201	245
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38	100	165	204	252
44	101	167a	209	253
45	102	168	210	255
46	104	169	211	256
49	108	170	212	258a
50	109	171	213	265
51	115a	172	215	266
53	118	173	216	267
54	119	174	218	268
55	121	175	219	272
58	122	176	221	273
60	124	178	222	274
61a	126	179	224	275
62	127	180b	225	276
63	130	181	230	277
64	143	183	231	280
65	144	185	232	281
66	150	186	235	295
67	153	187	236	296
71	154	192	239	339
76	155	193	241	341
77	158	194	242	355a

secondary containment penetration flow path not isolated.

(e) If Specification 3.1.1a(1)(a) is not met while in the refueling condition, then:

Immediately suspend core alterations, except for fuel assembly removal, and

Immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

(2) Reactivity margin - stuck control rods

Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. Inoperable control rods shall be valved out of service, in such positions that Specification 3.1.1a(1)(a) is met. In no case shall the number of non-fully inserted rods valved out of service be greater than six during power operation. If this specification is not met, the reactor shall be placed in the cold shutdown condition. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing.

INSERT 1

(2) Reactivity margin - stuck control rods

Each withdrawn control rod shall be exercised ~~at a frequency of 31 days~~ after the control rod has been withdrawn and power level is greater than the low power set point of the RWM. Insert each withdrawn control rod at least one notch.

This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

LIMITING CONDITION FOR OPERATION

c. Scram Insertion Times

(1) The average scram insertion time of all operable control rods, in the power operation condition, shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.00
90	5.00

(2) Except as noted in 3.1.1.c(3), the maximum insertion scram time, in the power operation condition, shall be no greater than:

% Inserted From Fully Withdrawn	Maximum Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.12
90	5.30

SURVEILLANCE REQUIREMENT

c. Scram Insertion Times

The maximum scram insertion time shall be demonstrated through measurement for*:

(1) All control rods prior to thermal power exceeding 40% power with reactor pressure above 800 psig, after each major refueling outage or after a reactor shutdown that is greater than 120 days.

(2) Specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods with reactor pressure above 800 psig.

INSERT 1

(3) At least 20 control ~~rods~~, on a rotating basis, on a frequency ~~of less than or equal to once per 180 days of cumulative power operation,~~ with reactor pressure above 800 psig.

* For single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(3) Control rods with longer scram insertion time will be permitted provided that no other control rod in a nine-rod square array around this rod has a: (a) Scram insertion time greater than the maximum allowed, (b) Malfunctioned accumulator, (c) Valved out of service in a non-fully inserted position.	INSERT 1
d. Control Rod Accumulators At all reactor operating pressures, a rod accumulator may be out of service provided that no other control rod in a nine-rod square array around this rod has a: (1) Malfunctioned accumulator, (2) Valved out of service in a non-fully inserted position, (3) Scram insertion greater than maximum permissible insertion time.	d. Control Rod Accumulators ~~Once a shift~~ check the status of the accumulator pressure and level alarms in the control room.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
If a control rod with a malfunctioned accumulator is inserted "full-in" and valved out of service, it shall not be considered to have a malfunctioned accumulator.	
e. Scram Discharge Volume	e. Scram Discharge Volume (SDV)
With one scram discharge volume vent valve and/or one scram discharge volume drain valve inoperable and open, restore the inoperable valve(s) to OPERABLE status within 24 hours.	Scram Discharge Volume Vent and Drain Valves shall be demonstrated OPERABLE during Power Operations by: INSERT 1 1. ~~At least once per month~~ verifying each valve to be open;* INSERT 1
With any scram discharge volume vent valve(s) and/or any scram discharge volume drain valve(s) otherwise inoperable, restore at least one vent and one drain valve to OPERABLE status within 8 hours.	2. ~~At least once per quarter~~ cycling each valve through at least one complete cycle of full travel; and
	The Scram Discharge Volume Drain and Vent valves shall be demonstrated OPERABLE ~~at least once per Operating Cycle~~ INSERT 1 by verifying that: 1. Valves close within 10 seconds after receipt of a signal for control rods to scram; 2. Valves open when the scram signal is reset; 3. Level instrumentation response proves that no blockage in the system exists.
	* These valves may be closed intermittently for testing under administrative controls.

The allowable inoperable rod patterns will be determined using information obtained in the startup test program supplemented by calculations. During initial startup, the reactivity condition of the as-built core will be determined. Also, sub-critical patterns of widely separated withdrawn control rods will be observed in the control rod sequences being used. The observations, together with calculated strengths of the strongest control rods in these patterns will comprise a set of allowable separations of malfunctioning rods. During the fuel cycle, similar observations made during any cold shutdown can be used to update and/or increase the allowable patterns.

The number of rods permitted to be valved out of service could be many more than the six allowed by the specification, particularly late in the operating cycle; however, the occurrence of more than six could be indicative of a generic problem and the reactor will be shut down. Placing the reactor in the shutdown condition inserts the control rods and accomplishes the objective of the specifications on control rod operability. This operation is normally expected to be accomplished within ten hours.

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures that the control rod is not stuck and is free to insert on a scram signal. This surveillance is not required when thermal power is less than or equal to the low power set point (LPSP) of the RWM, since notch insertion may not be compatible with the requirements of the RWM. ~~The 31 day surveillance test frequency takes into account operating experience related to changes in CRD performance. This surveillance requirement allows 31 days after withdrawal of the control rod concurrent with thermal power greater than the LPSP of the RWM to perform the surveillance.~~ INSERT 3

The requirement to exercise control rods at least once per 24 hours in the event power operation is continuing with two or more control rods which are valved out of service or one fully or partially withdrawn control rod which can not be moved, provides a reasonable time to test the control rods and provide assurance of the reliability of the remaining control rods.

b. Control Rod Withdrawal

(1) Control rod dropout accidents as discussed in Appendix E* can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides an indirect verification that the rod is coupled to its drive. Details of the control rod drive coupling are given in Section IV.B.6.1*.

*FSAR

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.1.2 LIQUID POISON SYSTEM

Applicability:

Applies to the operating status of the liquid poison system.

Objective:

To assure the capability of the liquid poison system to function as an independent reactivity control mechanism and as a post-LOCA suppression pool pH control mechanism.

Specification:

a. During power operating conditions, and whenever the reactor coolant system temperature is greater than 212°F except for reactor vessel hydrostatic or leakage testing with the reactor not critical, the liquid poison system shall be operable except as specified in 3.1.2.b.

b. If a redundant component becomes inoperable, Specification 3.1.2.a shall be considered fulfilled, provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.

4.1.2 LIQUID POISON SYSTEM

Applicability:

Applies to the periodic testing requirements for the liquid poison system.

Objective:

To specify the tests required to assure the capability of the liquid poison system for controlling core reactivity.

Specification:

The liquid poison system surveillance shall be performed as indicated below:

a. Overall System Test:

(1) ~~At least once during each operating cycle -~~

INSERT 1

Manually initiate the system from the control room. Demineralized water shall be pumped to the reactor vessel to verify minimum flow rates and demonstrate that valves and nozzles are not clogged.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. The liquid poison tank shall contain a minimum of 1325 gallons of boron bearing solution. The solution shall have a sufficient concentration of sodium pentaborate enriched with Boron-10 isotope to satisfy the equivalency equation.

$$\frac{C}{13\%\ wt} \times \frac{628300}{M} \times \frac{Q}{86\ GPM} \times \frac{E}{19.8\%\ Atom} \geq 1$$

Where:

C = Sodium Pentaborate Solution Concentration (Wt %)

M = Mass of Water in Reactor Vessel and Recirculation piping at Hot Rated Conditions (501500 lb)

Q = Liquid Poison Pump Flow Rate (30 GPM nominal)

E = Boron-10 Enrichment (Atom %)

d. The liquid poison solution temperature shall not be less than the temperature presented in Figure 3.1.2.b.

e. If Specifications "a" through "d" are not met, initiate normal orderly shutdown within one hour.

Remove the squibs from the valves and verify that no deterioration has occurred by actual field firing of the removed squibs. In addition, field fire one squib from the batch of replacements.

Disassemble and inspect the squib-operated valves to verify that valve deterioration has not occurred.

INSERT 1

(2) ~~At least once per month -~~

Demineralized water shall be recycled to the test tank. Pump discharge pressure and minimum flow rate shall be verified.

b. Boron Solution Checks:

(1) ~~At least once per month -~~

INSERT 1

Boron concentration shall be determined.

(2) ~~At least once per day -~~

INSERT 1

Solution volume shall be checked. In addition, the sodium pentaborate concentration shall be determined and conformance with the requirements of the equivalency equation shall be checked any time water or boron are added or if the solution temperature drops below the limits specified by Figure 3.1.2.b.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	(3) ~~At least once per day -~~ INSERT 1 The solution temperature shall be checked. (4) ~~At least once per operating cycle -~~ INSERT 1 Verify enrichment by analysis. c. <u>Surveillance with Inoperable Components</u> When a component becomes inoperable its redundant component shall be verified to be operable immediately and ~~daily~~ INSERT 1 thereafter.

The liquid poison system also has a post-LOCA safety function to buffer the suppression pool pH in order to maintain the bulk pH above 7.0. This function is necessary to prevent iodine re-evolution consistent with the Alternative Source Term analysis methodology. Manual system initiation is used, and the minimum amount of sodium pentaborate solution required to be injected for suppression pool pH buffering is 1114 gallons at a minimum concentration of 9.423 weight percent. This volume consists of the minimum required volume of 1325 gallons minus the 197 gallons that are contained below the point where the pump takes suction from the storage tank and minus 14 gallons that are assumed to remain in the pump suction and discharge piping after injection stops. Operation of a single liquid poison pump can satisfy this post-LOCA function. This function applies to the power operating condition, and also whenever the reactor coolant system temperature is greater than 212°F except for reactor vessel hydrostatic or leakage testing with the reactor not critical.

Specification 3.1.2.e requires initiation of a normal orderly plant shutdown within one hour if Specifications 3.1.2.a through 3.1.2.d are not met. Specifically, the plant must be brought to a reactor operating condition in which the LCO does not apply. To achieve this status, the reactor coolant system temperature must be reduced to ≤ 212°F by initiating a normal orderly shutdown using the normal plant shutdown procedure. Based on operating experience, the use of the normal plant shutdown procedure to achieve the plant shutdown results in a reasonable time to reach the required plant conditions from full power operating conditions in an orderly manner and without challenging plant systems.

Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than 7 days. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to 6 months.

The system test specified demonstrates component response such as pump starting upon manual system initiation and is similar to the operating requirement under accident conditions. The only difference is that demineralized water rather than the boron solution will be pumped to the reactor vessel. ~~The test interval between operating cycles results in a system failure probability of 1.1 x 10^-6 (Fifth Supplement, p. 115)* and is consistent with practical considerations.~~ INSERT 3

Pump operability will be demonstrated on a more frequent basis. A continuity check of the firing circuit on the explosive valves is provided by pilot lights in the control room. Tank level and temperature alarms are provided to alert the operator of off-normal conditions.

The functional test and other surveillance on components, along with the monitoring instrumentation, gives a high reliability for liquid poison system operability.

*FSAR

LIMITING CONDITION FOR OPERATION

3.1.3 EMERGENCY COOLING SYSTEM

Applicability:

Applies to the operating status of the emergency cooling system.

Objective:

To assure the capability of the emergency cooling system to cool the reactor coolant in the event the normal reactor heat sink is not available.

Specification:

a. During power operating conditions and whenever the reactor coolant temperature is greater than 212°F except for hydrostatic testing with the reactor not critical, both emergency cooling systems shall be operable except as specified in 3.1.3.b.

b. If one emergency cooling system becomes inoperable, Specification 3.1.3.a shall be considered fulfilled, provided that the inoperable system is returned to an operable condition within 7 days and the additional surveillance required in 4.1.3.f is performed.

SURVEILLANCE REQUIREMENT

4.1.3 EMERGENCY COOLING SYSTEM

Applicability:

Applies to periodic testing requirements for the emergency cooling system.

Objective:

To assure the capability of the emergency cooling system for cooling of the reactor coolant.

Specification:

The emergency cooling system surveillance shall be performed as indicated below:

INSERT 1

a. ~~At least once every five years~~ -

The system heat removal capability shall be determined.

INSERT 1

b. ~~At least once daily~~ -

The shell side water level and makeup tank water level shall be checked.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. Make up water shall be available from the two gravity feed makeup Water tanks.	c. ~~At least once per month~~ - INSERT 1 The makeup tank level control valve shall be manually opened and closed.
d. During Power Operating Conditions, each emergency cooling system high point vent to torus shall be operable. 1. With a vent path for one emergency cooling system inoperable, restore the vent path to an operable condition within 30 days. 2. With vent paths for both emergency cooling systems inoperable, restore one vent path to an operable condition with 14 days and both vent paths within 30 days.	d. ~~At least once each shift~~ - INSERT 1 The area temperature shall be checked.
e. If Specification 3.1.3.a, b, c, or d are not met, a normal orderly shutdown shall be initiated within one hour, and the reactor shall be in the cold shutdown conditions within ten hours.	e. ~~During each major refueling outage~~ - INSERT 1 Automatic actuation and functional system testing shall be performed ~~during each major refueling outage~~ INSERT 1 and whenever major repairs are completed on the system. Each emergency cooling vent path shall be demonstrated operable by cycling each power-operated valve (05-01R, 05-11, 05-12, 05-04R, 05-05 and 05-07) in the vent path through one complete cycle of full travel and verifying that all manual valves are in the open position. f. Surveillance with an Inoperable System - When one of the emergency cooling systems is inoperable, the level control valve and the motor-operated isolation valve in the operable system shall be verified to be operable immediately and ~~daily~~ INSERT 1 thereafter.

Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than 7 days. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to 6 months.

~~The system heat removal capability shall be determined at five year intervals. This is based primarily on the low corrosion characteristics of the stainless steel tubing. During normal plant operation the water level will be observed at least once daily on emergency condensers and makeup water tanks. High and low water level alarms are also provided on the above pieces of equipment. The test frequency selected for level checks and valve operation is to assure the reliability of the system to operate when required.~~

The emergency cooling system is provided with high point vents to exhaust noncondensible gases that could inhibit natural circulation cooling. Valve redundancy in the vent path serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path. The function, capabilities and testing requirements of the emergency cooling vent paths are consistent with the requirements of item II.B.1 of NUREG 0737, "Clarification of TMI Action Plan Requirement," November 1980.

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.1.4 CORE SPRAY SYSTEM

Applicability:

Applies to the operating status of the core spray systems.

Objective:

To assure the capability of the core spray systems to cool reactor fuel in the event of a loss-of-coolant accident.

Specification:

a. Whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, each of the two core spray systems shall be operable except as specified in Specifications b and c below.

b. If a redundant component of a core spray system becomes inoperable, that system shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.

c. If a redundant component in each of the core spray systems becomes inoperable, both systems shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.

4.1.4 CORE SPRAY SYSTEM

Applicability

Applies to the periodic testing requirements for the core spray systems.

Objective:

To verify the operability of the core spray systems.

Specification:

The core spray system surveillance shall be performed as indicated below.

INSERT 1

a. ~~At each major refueling outage~~ automatic actuation of each subsystem in each core spray system shall be demonstrated.

INSERT 1

b. ~~At least once per quarter~~ pump operability shall be checked.

INSERT 1

c. ~~At least once per quarter~~ the operability of power-operated valves required for proper system operation shall be checked.

LIMITING CONDITION FOR OPERATION

d. If Specifications a, b and c are not met, a normal orderly shutdown shall be initiated within one hour and the reactor shall be in the cold shutdown condition within ten hours.

e. During reactor operation, except during core spray system surveillance testing, core spray isolation valves 40-02 and 40-12 shall be in the open position and the associated valve motor starter circuit breakers for these valves shall be locked in the off position. In addition, redundant valve position indication shall be available in the control room.

f. Whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F, two core spray subsystems shall be operable except as specified in g and h below.

g. If one of the above required subsystems becomes inoperable, restore at least two subsystems to an operable status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENT

d. (Deleted)

e. Surveillance with Inoperable Components

When a component becomes inoperable its redundant component or system shall be verified to be operable immediately and ~~daily~~ thereafter.

INSERT 1

f. With a core spray subsystem suction from the CST, CST level shall be checked ~~once per day~~.

INSERT 1

INSERT 1

g. ~~At least once per month~~ when the reactor coolant temperature is greater than 212°F, verify that the piping system between valves 40-03, 13 and 40-01, 09, 10, 11 is filled with water.

The testing specified for each major refueling outage will demonstrate component response upon automatic system initiation. For example, pump set starting (low-low level or high drywell pressure) and valve opening (low-low level or high drywell pressure and low reactor pressure) must function, under simulated conditions, in the same manner as the systems are required to operate under actual conditions. The only differences will be that demineralized water rather than suppression chamber water will be pumped to the reactor vessel and the reactor will be at atmospheric pressure. The core spray systems are designed such that demineralized water is available to the suction of one set of pumps in each system (Section VII-Figure VII-1)*.

~~The system test interval between operating cycles results in a system failure probability of 1.1 x 10^-6 (Fifth Supplement, page 115) and is consistent with practical considerations. The more frequent component testing results in a more reliable system.~~ INSERT 3

~~At quarterly intervals,~~ startup of core spray pumps will demonstrate pump starting and operability. No flow will take place to the reactor vessel due to the lack of a low-pressure permissive signal required for opening of the blocking valves. A flow restricting device has been provided in the test loop which will create a low pressure loss for testing of the system. In addition, the normally closed power operated blocking valves will be manually opened and re-closed to demonstrate operability.

The intent of Specification 3.1.4i is to allow core spray operability at the time that the suppression chamber is dewatered which will allow normal refueling activities to be performed. With a core spray pump taking suction from the CST, sufficient time is available to manually initiate one of the two raw water pumps that provide an alternate core spray supply using lake water. Both raw water pumps shall be operable in the event the suppression chamber was dewatered.

INSERT 1

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.1.5 SOLENOID-ACTUATED PRESSURE RELIEF VALVES (AUTOMATIC DEPRESSURIZATION SYSTEM)	4.1.5 SOLENOID-ACTUATED PRESSURE RELIEF VALVES (AUTOMATIC DEPRESSURIZATION SYSTEM)
Applicability:	Applicability:
Applies to the operational status of the solenoid-actuated relief valves.	Applies to the periodic testing requirements for the solenoid-actuated pressure relief valves.
Objective:	Objective:
To assure the capability of the solenoid-actuated pressure relief valves to provide a means of depressurizing the reactor in the event of a small line break to allow full flow of the core spray system.	To assure the operability of the solenoid-actuated pressure relief valves to perform their intended functions.
Specification:	Specification:
a. During power operating condition whenever the reactor coolant pressure is greater than 110 psig and the reactor coolant temperature is greater than saturation temperature, all six solenoid-actuated pressure relief valves shall be operable.	The solenoid-actuated pressure relief valve surveillance shall be performed as indicated below. INSERT 1 a. ~~At least once during each operating cycle~~, verify each valve actuator strokes when manually actuated.
b. If specification 3.1.5a above is not met, the reactor coolant pressure and the reactor coolant temperature shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.	INSERT 1 b. ~~At least once during each operating cycle,~~ automatic initiation shall be demonstrated.

Pilot Valve – Maintenance is performed on the pilot valve assemblies during each refueling outage which includes a range of mechanical inspection and leak rate testing criteria. The inspection and maintenance activities include leak rate testing and demonstration of pilot stem/disc freedom of movement and reseat functionality. Following installation of the pilot valve assembly inside the housing, the pilot valve operating lever and pilot valve assembly freedom of movement and clearance adjustments are confirmed, followed by stroking the solenoid actuator plunger by hand to the full extent of travel. This ensures that the solenoid actuator plunger, pilot valve operating lever, and pilot valve assembly function as a unit.

Main Valve – Main valve testing will be performed at an offsite steam test facility. A spare pilot valve assembly and solenoid actuator, both representative of the components used at the plant, will be installed at the test facility to allow testing the main valve. The valve will be installed on a test steam header in the same orientation as the plant installation. The test conditions in the test facility will be similar to those in the plant, including ambient temperature and steam conditions. The main valve will receive an initial seat leak test, a functional test to ensure it is capable of opening and closing, and a final seat leak test. Valve stroke time will be obtained during the exercise test. Valve seat tightness will be verified by a cold bar test, and if not free of fog, leakage will be measured and verified to be below specified acceptance criteria. After initial testing the main valves will be completely disassembled, inspected and refurbished, and then retested. The refurbished main valves will be stored at the offsite test facility and returned to the plant prior to the next scheduled use.

2. The second method is a manual actuation of the relief valve with the reactor at nominal operating pressure. Verification that the relief valve has opened and steam is flowing from the valve is provided by the response of the turbine bypass valves, a change in the measured steam flow, and indications on the acoustic monitor or thermocouple downstream of the valve. Manual actuation of the solenoid actuator during this test or during the maintenance procedures performed prior to this test will satisfy the surveillance requirement. Adequate reactor steam dome pressure must be available to perform the steam test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or main turbine bypass valves to continue to control reactor pressure when the solenoid-actuated relief valve diverts steam flow upon opening. Steam from the reactor vessel is discharged to the suppression pool during valve testing. The malfunction analysis (Section II.XV of the "Technical Supplement to Petition to Increase Power Level," dated April 1970) demonstrates that no serious consequences result if one valve fails to close, as no vessel or fuel stress limits are approached. Testing the relief valves with the reactor at nominal operating pressure assures that the relief valves can operate under nominal operating conditions. During reactor startup, while reactor pressure is increasing before the valves are tested, relief valve operability has not been positively verified. This is acceptable because: (1) redundant safety systems are provided to ensure adequate core cooling in the event of a small break LOCA with loss of feedwater and multiple relief valve failures; and (2) dynamic loads and suppression pool heatups associated with high pressure testing are within allowable limits.

INSERT 3

LIMITING CONDITION FOR OPERATION

3.1.6 CONTROL ROD DRIVE PUMP COOLANT INJECTION

Applicability:

Applies to the operational status of the control rod drive pump coolant injection system.

Objective:

To assure the capability of the control rod drive pump coolant injection system to:

Provide core cooling in the event of a small line break, and

Provide coolant makeup in the event of reactor coolant leakage (see LCO 3.2.5).

Specification:

a. Whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, the control rod drive pump coolant injection system shall be operable except as specified in "b" below.

SURVEILLANCE REQUIREMENT

4.1.6 CONTROL ROD DRIVE PUMP COOLANT INJECTION

Applicability:

Applies to the periodic testing requirements for the control rod drive pump coolant injection system.

Objective:

To assure the capability of the control rod drive pump coolant injection system in performing its intended functions.

Specification:

The control rod drive pump coolant injection system surveillance shall be performed as indicated below.

INSERT 1

a. ~~At least once per operating cycle~~ -

Automatic starting of each pump shall be demonstrated.

LIMITING CONDITION FOR OPERATION

b. If a redundant component becomes inoperable, the control rod drive pump coolant injection system shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.

c. If Specifications "a" or "b" above are not met, the reactor coolant temperature shall be reduced to 212°F or less within ten hours.

SURVEILLANCE REQUIREMENT

b. ~~At least once per quarter -~~

INSERT 1

Pump flow rate shall be determined.

c. Surveillance with Inoperable Components

When a component becomes inoperable, its redundant component shall be verified to be operable immediately and ~~daily~~ thereafter.

INSERT 1

BASES FOR 3.1.6 AND 4.1.6 CONTROL ROD DRIVE PUMP COOLANT INJECTION

The high pressure coolant injection capability of the control rod drive pumps is used to provide high pressure makeup for the specified leakage of 25 gpm (see LCO 3.2.5) and to provide core cooling in the case of a small line break. Each pump can supply 50 gpm water makeup to the reactor vessel.

One pump will normally be operating. Electric power for this system is normally available from the reserve transformer. Automatic initiation is provided to start each pump on its respective diesel generator in case offsite power is lost.

The system minimum delivery rate of 50 gpm within 60 seconds of receipt of signal will assure that automatic pressure blowdown is not actuated for the specified leakage rate of 25 gpm.

The 60-second delay in pump starting is acceptable since at least 15 minutes are available before the triple low reactor water level signals the automatic pressure blowdown to start. This analysis was based on the following assumptions; no makeup to the reactor vessel, a 50 gpm (two times allowable) leak rate exists, and the emergency condensers over-perform by 10 percent.

Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than 7 days. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to 6 months.

~~The testing specified during an operating cycle will demonstrate component response upon automatic system initiation in the same manner that the system will operate if required. The testing interval results in a calculated failure probability of 1.1 x 10^-6 for a control rod drive pump (Fifth Supplement), and is compatible with practical considerations. Continual monitoring of pump performance is provided since one pump is normally operating and instrumentation and alarms monitor operation of flow and pressure regulation (Section X)*~~.

INSERT 3

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.1.7 FUEL RODS	4.1.7 FUEL RODS

3.1.7 FUEL RODS

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall not exceed the limiting value provided in the Core Operating Limits Report. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) shown in the Core Operating Limits Report. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded at any node in the core, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR at all nodes in the core is

4.1.7 FUEL RODS

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specification:

a. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of axial location and average planar exposure shall be determined ~~daily~~ INSERT 1 during reactor operation at ≥25 percent rated thermal power.

LIMITING CONDITION FOR OPERATION

not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until APLHGR at all nodes is within the prescribed limits.

b. Linear Heat Generation Rate (LHGR)

During power operation, the Linear Heat Generation Rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the limiting value specified in the Core Operating Limits Report.

If at any time during power operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded at any location, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR at all locations is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until LHGR at all locations is within the prescribed limits.

c. Minimum Critical Power Ratio (MCPR)

During power operation, the MCPR for all fuel at rated power and flow shall be within the limit provided in the Core Operating Limits Report.

If at any time during power operation it is determined by normal surveillance that the above limit is no longer met, action shall be initiated within 15 minutes to restore operation to within

SURVEILLANCE REQUIREMENT

b. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked ~~daily~~ during reactor operation at ≥25% rated thermal power.

INSERT 1

c. Minimum Critical Power Ratio (MCPR) INSERT 1

(1) MCPR shall be determined ~~daily~~ during reactor power operation at >25% rated thermal power.

(2) MCPR operating limit shall be determined within 72 hours of completing scram time testing as required in Specification 4.1.1(c).

LIMITING CONDITION FOR OPERATION

the prescribed limit. If all the operating MCPRs are not returned to within the prescribed limit within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until MCPR is within the prescribed limit. For core flows other than rated, the MCPR limit shall be the limit identified above times K_f where K_f is provided in the Core Operating Limits Report.

d. Power Flow Relationship During Operation

This power/flow relationship shall not exceed the limiting values shown in the Core Operating Limits Report.

If at any time during power operation it is determined by normal surveillance that the limiting value for the power/flow relationship is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the power/flow relationship is not returned to within the prescribed limits within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until the power/flow relationship is within the prescribed limits.

e. Partial Loop Operation

During power operation, partial loop operation is permitted provided the following conditions are met.

SURVEILLANCE REQUIREMENT

d. Power Flow Relationship

Compliance with the power flow relationship in Section 3.1.7.d shall be determined ~~daily~~ during reactor operation.

INSERT 1

e. Partial Loop Operation

Under partial loop operation, surveillance requirements 4.1.7, a, b, c and d above are applicable.

Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature and the peak local cladding oxidation following the postulated design basis loss-of-coolant accident will not exceed the limits specified in 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ±20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50, Appendix K limit. The limiting value for APLHGR is provided in the Core Operating Limits Report. The APLHGR curves in the Core Operating Limits Report are based on calculations using the models described in Reference 15.

The LOCA analyses are sensitive to minimum critical power ratio (MCPR). In the Reference 15 analysis, an MCPR value of 1.30 was assumed. If future transient analyses should yield a MCPR limit below this value, the Reference 15 LOCA analysis MCPR value would become limiting. The current MCPR limit is provided in the Core Operating Limits Report.

Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated (Reference 12). The LHGR shall be checked ~~daily~~ periodically during reactor operation at ≥25% power to determine if fuel burnup or control rod movement has caused changes in power distribution.

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.1.8 HIGH PRESSURE COOLANT INJECTION

Applicability:

Applies to the operational status of the high pressure coolant injection system.

Objective:

To assure the capability of the high pressure coolant injection system to cool reactor fuel in the event of a loss-of-coolant accident.

Specification:

a. During the power operating condition* whenever the reactor coolant pressure is greater than 110 psig and the reactor coolant temperature is greater than saturation temperature, the high pressure coolant injection system shall be operable except as specified in Specification "b" below.

b. If a redundant component of the high pressure coolant injection system becomes inoperable, the high pressure coolant injection shall be considered operable provided that the component is returned to an operable condition within 15 days and the additional surveillance required is performed.

* One Feedwater Pump blocking valve in one HPCI pump train may be closed during reactor startup when core power is equal to or less than 25% of rated thermal power.

4.1.8 HIGH PRESSURE COOLANT INJECTION

Applicability:

Applies to the periodic testing requirements for the high pressure coolant injection system.

Objective:

To verify the operability of the high pressure coolant injection system.

Specification:

The high pressure coolant injection surveillance shall be performed as indicated below:

INSERT 1

a. ~~At least once per operating cycle~~ -

Automatic start-up of the high pressure coolant injection system shall be demonstrated.

INSERT 1

b. ~~At least once per quarter~~ -

Pump operability shall be determined.

LIMITING CONDITION FOR OPERATION

c. If Specification "a" and "b" are not met, a normal orderly shutdown shall be initiated within one hour and reactor coolant pressure and temperature shall be reduced to less than 110 psig and saturation temperature within 24 hours.

SURVEILLANCE REQUIREMENT

c. Surveillance with Inoperable Components

When a component becomes inoperable, its redundant component shall be verified to be operable immediately and ~~daily~~ thereafter.

INSERT 1

The High Pressure Coolant Injection System (HPCI) is provided to ensure adequate core cooling in the unlikely event of small reactor coolant line break. The HPCI System is available for line breaks which exceed the capability of the Control Rod Drive pumps and which are not large enough to allow fast enough depressurization for core spray to be effective.

One set of high pressure coolant injection pumps consists of a condensate pump, a feedwater booster pump and a motor driven feedwater pump. One set of pumps is capable of delivering 3,420 gpm to the reactor vessel at reactor pressure. The performance capability of HPCI alone and in conjunction with other systems to provide adequate core cooling for a spectrum of line breaks is discussed in the Fifth Supplement of the FSAR.

In determining the operability of the HPCI system, the required performance capability of various components shall be considered.

a. The HPCI System shall be capable of meeting at least 3,420 gpm flow at normal reactor operating pressure.

b. The motor driven feedwater pump shall be capable of automatic initiation upon receipt of either an automatic turbine trip signal or reactor low-water-level signal.

c. The Condenser hotwell level shall not be less than 57 inches (75,000 gallons).

d. The Condensate storage tanks inventory shall not be less than 105,000 gallons.

e. The motor-driven feedwater pump will automatically trip if reactor high water level is sustained for ten seconds and the associated pump downstream flow control valve is not closed.

During reactor startup and shutdown, only the condensate and feedwater booster pumps are in operation at reactor pressures below approximately 400 psig. The feedwater pump is in standby. However, if the HPCI initiation signal occurs, the feedwater pump would automatically start. Calculations show that the condensate and feedwater booster pump alone are capable of providing 3,420 gpm at a reactor pressure of approximately 270 psig.

The capability of the condensate, feedwater booster and motor driven feedwater pumps will be demonstrated by their operation as part of the feedwater supply during normal station operation. ~~Stand-by pumps will be placed in service at least quarterly to supply feedwater during station operation. An automatic system initiation test will be performed at least once per operating cycle. This will involve automatic starting of the motor driven feedwater pumps and flow to the reactor vessel.~~

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.2.3 COOLANT CHEMISTRY

Applicability:

Applies to the reactor coolant system chemical requirements.

Objective:

To assure the chemical purity of the reactor coolant water.

Specification:

a. The reactor coolant water shall not exceed the following limits for > 24 hours with the coolant temperature ≥200 degrees F and reactor thermal power ≤ 10%, or a shutdown shall be initiated within 1 hour and the reactor shall be shutdown and reactor coolant temperature be reduced to <200 degrees F within 10 hours.

Conductivity	1 μmho/cm*
Chloride ion	100 ppb
Sulfate ion	100 ppb

* During Noble Metal Chemical Addition (NMCA), the limit is 20 μmho/cm. Post NMCA, the conductivity limit is 2 μmho/cm for up to a 5 month period at power operation.

4.2.3 COOLANT CHEMISTRY

Applicability:

Applies to the periodic testing requirements of the reactor coolant chemistry.

Objective:

To determine the chemical purity of the reactor coolant water.

Specification:

INSERT 1

Samples shall be taken and analyzed for conductivity, chloride and sulfate ion content ~~daily~~. In addition, if the conductivity becomes abnormal (other than short term spikes) as indicated by the continuous conductivity monitor, samples shall be taken and analyzed within 8 hours.

When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken and analyzed for conductivity, chloride and sulfate ion content at least once per 8 hours.

In its May 8, 1997 letter, the NRC required that the licensee submit an application for amendment to address the differences between the current TS conductivity limits for reactor coolant chemistry and the analysis assumptions for the core shroud crack growth evaluations. The purpose of this specification is to limit intergranular stress corrosion cracking (IGSCC) crack growth rates through the control of reactor coolant chemistry. The LCO values ensure that transient conditions are acted on to restore reactor coolant chemistry values to normal in a reasonable time frame. Under transient conditions, potential crack growth rates could exceed analytical assumptions, however, the duration will be limited so that any effect on potential crack growth is minimized and the design basis assumptions are maintained. The plant is normally operated such that the average coolant chemistry for the operating cycle is maintained at the conservative values of < 0.19 μmho/cm for conductivity and < 5 ppb for chloride ions and < 5 ppb for sulfate ions. This will ensure that the crack growth rate is bounded by the core shroud analysis assumptions. Since these are average values, there are no specific LCO actions to be taken if these values are exceeded at a specific point in time. The EPRI "BWR Water Chemistry Guidelines-1996 Revision" (EPRI TR-103515-R1, BWRVIP-29) action level 1 guidelines suggest that if conductivity is above 0.3 μS/cm, or chloride or sulfate ions exceed 5 ppb, that corrective action be initiated as soon as possible and to restore levels below level 1 within 96 hours. If the parameters are not reduced to below these levels within 96 hours, complete a review and implement a program and schedule for implementing corrective measures.

Specifications 3.2.3a, b, and c are consistent with the licensee's commitment to Table 4.4 of the BWR water chemistry guidelines. The 24 hour action time period for exceeding the coolant chemistry limits described in 3.2.3a and b ensures that prompt action is taken to restore coolant chemistry to normal operating levels. The requirement to commence a shutdown within 1 hour, and to be shutdown and reactor coolant temperature be reduced to < 200 degrees F within 10 hours minimizes the potential for IGSCC crack growth. Reactor water samples are analyzed daily to ensure that reactor water quality remains within the BWR water chemistry guidelines. These samples are analyzed and compared to action level 1 values.

The conductivity of the reactor coolant is periodically ~~continuously~~ monitored, and ~~. The continuous conductivity monitor~~ is visually checked ~~shiftly~~ in accordance with procedures. The monitor alarms at the local panel. The recorder, which is located in the Control Room, alarms in the Control Room. The samples of the coolant which are analyzed for conductivity ~~daily~~ will serve as a comparison with the continuous conductivity monitor. The primary sample point for the reactor water conductivity samples is the non-regenerative heat exchanger in the reactor water cleanup system. An alternate sample point is the #11 recirculation loop. The reactor coolant samples will also be used to determine the chloride and sulfate concentrations. ~~Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride and sulfate ion content.~~ However, if the conductivity becomes abnormal (>0.19 μmho/cm), other than short term spikes, chloride and sulfate measurements will be made within 8 hours to assure that the normal limits (< 5 ppb of chloride or sulfate ions) are maintained. A short term spike is defined as a rise in conductivity (> 0.19 μmho/cm) such as that which could arise from injection of additional feedwater flow for a duration of approximately 30 minutes in time. These actions will minimize the potential for IGSCC crack growth. INSERT 3

NMP1 will use Noble Metal Chemical Addition (NMCA) as a method to enhance the effectiveness of Hydrogen Water Chemistry (HWC) in mitigating IGSCC. NMCA will result in temporary increases in reactor coolant conductivity values during and following application. During application, the conductivity limit specified in 3.2.3a and 3.2.3c.1 is increased to 20 μmho/cm. The application period includes post-NMCA injection cleanup activities conducted prior to returning the plant to power operation. An increase in conductivity is expected principally due to residual ionic species from the NMCA. However, these species have minor effects on IGSCC and are, therefore, acceptable. During NMCA, samples will be obtained from the temporary skid which is placed in service during the NMCA injection process.

LIMITING CONDITION FOR OPERATION

3.2.4 REACTOR COOLANT SPECIFIC ACTIVITY

Applicability:

Applies to the limits on reactor coolant specific activity.

Objective:

To assure that in the event of a reactor coolant system line break outside the drywell permissible doses are not exceeded.

Specification:

a. During the power operating and hot shutdown conditions, the specific activity of the reactor coolant shall be limited to Dose Equivalent I-131 specific activity ≤ 0.2 μCi/gm.

b. If reactor coolant specific activity is > 0.2 μCi/gm and ≤ 4.0 μCi/gm Dose Equivalent I-131, determine the Dose Equivalent I-131 once per 4 hours and restore Dose Equivalent I-131 to within the limit of Specification 3.2.4.a within 48 hours.

SURVEILLANCE REQUIREMENT

4.2.4 REACTOR COOLANT SPECIFIC ACTIVITY

Applicability:

Applies to the periodic testing requirements of the reactor coolant specific activity.

Objective:

To assure that limits on coolant specific activity are not exceeded.

Specification:

a. When the unit is in the power operating condition, verify that reactor coolant Dose Equivalent I-131 specific activity is ≤ 0.2 μCi/gm ~~once per 7 days~~. INSERT 1

b. Verify that reactor coolant Dose Equivalent I-131 specific activity is ≤ 0.2 μCi/gm within 24 hours prior to raising the reactor coolant temperature > 215°F, with the reactor not critical, and with primary containment integrity not established.

BASES FOR 3.2.4 AND 4.2.4 REACTOR COOLANT SPECIFIC ACTIVITY

The specific activity in the reactor coolant is an initial condition for evaluation of the radiological consequences of a main steam line break (MSLB) outside of primary containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely. The specific iodine activity is limited to $\leq 0.2\ \mu Ci/gm$ Dose Equivalent I-131. This limit ensures that the source term assumed in the radiological consequences analysis for the MSLB accident is not exceeded, so that any release of radioactivity to the environment during a MSLB results in offsite and control room radiation doses that satisfy the acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183. It is also conservative with respect to the value used in the radiological consequences analyses for other postulated small break loss of coolant accidents outside of primary containment and for postulated instrument line breaks.

The limits on reactor coolant specific activity are applicable in the power operating and hot shutdown conditions, since there is an escape path for release of radioactive material from the reactor coolant system to the environment in the event of a MSLB outside of primary containment. In the cold shutdown, refueling, and major maintenance conditions, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

When the reactor coolant specific activity exceeds the limit of $0.2\ \mu Ci/gm$ Dose Equivalent I-131, but is $\leq 4.0\ \mu Ci/gm$, samples must be analyzed for Dose Equivalent I-131 at least once every 4 hours. In addition, the specific activity must be restored to the limit within 48 hours. The completion time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour completion time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

The isotopic analyses of reactor water samples will be used to assure that the limit of Specification 3.2.4 is not exceeded during normal operation. ~~The 7 day frequency~~ is adequate to trend changes in the iodine activity level. The surveillance requirement need only be performed during the power operating condition because the level of fission products generated in other operating conditions is much less. In addition, the trend of the stack offgas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant.

INSERT 3, and

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, as discussed in the bases for Specification 3.6.2, some capability to detect gross fuel element failures is inherent in the radiation monitors in the offgas system and on the main steam lines.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.2.5 REACTOR COOLANT SYSTEM LEAKAGE

Applicability:

Applies to the limits on reactor coolant system leakage rate and leakage detection systems.

Objective:

To assure that the makeup capability provided by the control rod drive pump is not exceeded.

Specification:

a. Any time irradiated fuel is in the reactor vessel and the reactor temperature is above 212°F, reactor coolant leakage into the primary containment shall be limited to:

1. Five gallons per minute unidentified leakage.

2. A two gallon per minute increase in unidentified leakage within any period of 24 hours or less.

3. Twenty-five gallons per minute total leakage (identified plus unidentified) averaged over any 24 hour period.

4.2.5 REACTOR COOLANT SYSTEM LEAKAGE

Applicability:

Applies to the monitoring of reactor coolant system leakage.

Objective:

To determine the reactor coolant system leakage rate and assure that the leakage limits are not exceeded.

Specification:

a. A check of the reactor coolant leakage shall be made ~~every four hours~~.

INSERT 1

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, at least one of the leakage measurement channels associated with each sump (one for the drywell floor drain and one for the equipment drain) shall be operable.	b. The following surveillance shall be performed on each leakage detection system: INSERT 1
If the conditions a or b cannot be met, the reactor will be placed in the cold shutdown condition within 24 hours.	(1) An instrument calibration ~~once each refueling outage.~~
	(2) An instrument functional test ~~once every three months.~~ INSERT 1

Another method of determining reactor coolant leakage rate is by monitoring for excess leakage in the drywell floor and equipment drain tanks. This system monitors the change in tank volume over accurate time periods for the full range of tank instrumentation. If the leakage is high enough, an alarm is actuated indicating a leak rate above the predetermined limit (Section V.B)*.

Additional information is available to the operator which can be used for the shift leakage check if the drywell sumps level alarms are out of service. The integrated flow pumped from the sumps to the waste disposal system can be checked.

Qualitative information is also available to the operator in the form of indication of drywell atmospheric conditions. Continuous leakage from the primary coolant system would cause an increase in drywell temperature. Any leakage in excess of 15 gpm of steam would cause a continuing increase in drywell pressure with resulting scram (First Supplement)*.

Either the rate of rise leak detection system, the excess leakage detection system or the integrated flow can be utilized to satisfy Specification 3.2.5.b.

INSERT 3

*FSAR

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.2.7 REACTOR COOLANT SYSTEM ISOLATION VALVES

Applicability:

Applies to the operating status of the system of isolation valves on lines connected to the reactor coolant system.

Objective:

To assure the capability of the reactor coolant system isolation valves to minimize reactor coolant loss in the event of a rupture of a line connected to the nuclear steam supply system, and to minimize potential leakage paths from the primary containment in the event of a loss-of-coolant accident.

Specification:

a. Whenever fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, all reactor coolant system isolation valves on lines connected to the reactor coolant system shall be operable except as specified in Specification 3.2.7.b below.

b. In the event any isolation valve becomes inoperable whenever fuel is in the reactor vessel and the reactor coolant temperature is greater than 212°F, the system shall be considered operable provided that within 4 hours at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition, except as noted in Specification 3.1.1.e.

4.2.7 REACTOR COOLANT SYSTEM ISOLATION VALVES

Applicability:

Applies to the periodic testing requirement for the reactor coolant system isolation valves.

Objective:

To assure the capability of the reactor coolant system isolation valves to minimize reactor coolant loss in the event of a rupture of a line connected to the nuclear steam supply system, and to limit potential leakage paths from the primary containment in the event of a loss-of-coolant accident.

Specification:

The reactor coolant system isolation valves surveillance shall be performed as indicated below.

INSERT 1

a. ~~At least once per operating cycle~~ the operable automatically initiated power-operated isolation valves shall be tested for automatic initiation and closure times.

b. Additional surveillances shall be performed as required by Specification 6.5.4.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. If Specifications 3.2.7a and b above are not met, initiate normal orderly shutdown within one hour and have reactor in the cold shutdown condition within ten hours.	c. ~~At least once per quarter~~ INSERT 1 the feedwater and main-steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.
d. Whenever fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F, the isolation valves on the shutdown cooling system lines connected to the reactor coolant system shall be operable except as specified in Specification 3.2.7.e below.	d. ~~At least once per plant cold shutdown~~ INSERT 1 the feedwater and main steam line power-operated isolation valves shall be fully closed and reopened, unless this test has been performed within the previous 92 days.
e. In the event any shutdown cooling system isolation valve becomes inoperable whenever fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F, the system shall be considered operable provided that, within 4 hours, at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.	
f. If Specifications 3.2.7.d and 3.2.7.e above are not met, either: (1) Immediately initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs); or (2) Immediately initiate action to restore the valve(s) to operable status.	

and outboard isolation valves. To prevent a spurious or inadvertent valve opening from defeating the water seal, the motor-operated shutdown cooling system isolation valves are required to be de-activated (power is removed) during normal plant operation. Thus, the motor-operated shutdown cooling system isolation valves are considered operable when the valves are closed and de-activated and the water seal is capable of performing its function.

When the shutdown cooling system is placed in service for plant cooldown (with reactor pressure ≤ 120 psig and temperature ≤ 350°F), power for the motor-operated isolation valves must be restored and the valves opened. Should a loss of coolant accident occur at this time, failure of an isolation valve to close upon receipt of an isolation signal could cause a loss of the water seal. The risk associated with this potential single failure has been determined to be acceptable based on the low probability of a core damage event occurring during shutdown cooling system operation[(2)].

Specification 3.2.7.d requires operability of the shutdown cooling system isolation valves whenever fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F. If any isolation valve becomes inoperable, Specification 3.2.7.e requires that, within 4 hours, at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition. However, if the shutdown cooling function is needed to provide core cooling, isolating the shutdown cooling line is not desirable. Specification 3.2.7.f allows the shutdown cooling line to remain unisolated provided action is immediately initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs). If suspending the OPDRVs would result in closing the shutdown cooling system isolation valves, an alternative action is provided to immediately initiate action to restore the valve(s) to operable status. This allows the shutdown cooling system to remain in service while actions are being taken to restore the valve(s). The term "immediately" means that the action should be pursued without delay and in a controlled manner. Either of the actions identified in Specification 3.2.7.f must continue until OPDRVs are suspended or the valves are restored to operable status. Operation with the shutdown cooling system in service is not considered an OPDRV so long as system integrity is maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system. In addition, with the reactor coolant temperature less than or equal to 212°F, the water seal function is not required to consider the shutdown cooling system isolation valves operable since primary containment integrity is not required with reactor coolant temperature less than or equal to 215°F.

~~The valve operability test intervals are based on periods not likely to significantly affect operations, and are consistent with testing of other systems.~~ *Results obtained during closure testing are not expected to differ appreciably from closure times under accident conditions as in most cases, flow* helps to seal the valve.

~~The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1 x 10^{-7} (Fifth Supplement, p. 115)[(3)] that a line will not isolate. Additional surveillances are in accordance with the Inservice Testing Program described in Specification 6.5.4.~~ INSERT 3

(2) Letter from G. E. Edison (NRC) to B. R. Sylvia (NMPC) dated March 20, 1995, Issuance of Amendment for Nine Mile Point Nuclear Station Unit No. 1 (License Amendment No. 154).

(3) FSAR

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.2.8 PRESSURE RELIEF SYSTEMS-SAFETY VALVES

Applicability:

Applies to the operational status of the safety valves.

Objective:

To assure the capability of the safety valves to limit reactor overpressure below the safety limit in the event of rapid reactor isolation and failure of all pressure relieving devices.

Specification:

a. During power operating conditions and whenever the reactor coolant pressure is greater than 110 psig and temperature greater than saturation temperature all nine of the safety valves shall be operable.

b. If specification 3.2.8a is not met, the reactor coolant pressure and temperature shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.

4.2.8 PRESSURE RELIEF SYSTEMS-SAFETY VALVES

Applicability:

Applies to the periodic testing requirements for the safety valves.

Objective:

To assure the capability of the safety valves to limit reactor overpressure to below the safety limit.

Specification:

INSERT 1

~~At least once during each operating cycle,~~ the number of safety valves as determined by the IST Program Plan shall be removed, tested for set point and partial lift, and then returned to operation or replaced.

BASES FOR 3.2.8 AND 4.2.8 PRESSURE RELIEF SYSTEM-SAFETY VALVES

The required number of operable safety valves is based on a condition of main steam isolation valve closure while operating at 1850 Mwt, followed by a reactor scram on high neutron flux. Operation of all 9 safety valves will limit reactor pressure below the safety limit of 1375 psig.

~~The safety valve testing and intervals between tests are based on manufacturer's recommendations and past experience with spring actuated safety valves.~~

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	INSERT 1 b. ~~At least once during each operating cycle,~~ verify each valve actuator strokes when manually actuated.
	INSERT 1 c. ~~At least once during each operating cycle,~~ relief valve setpoints shall be verified.

BASES FOR 3.2.9 AND 4.2.9 PRESSURE RELIEF SYSTEM - SOLENOID ACTUATED PRESSURE RELIEF VALVES

As discussed in 2.2.2 and 3.2.8 above, the solenoid-actuated pressure relief valves are used to avoid actuation of the safety valves. The set points of the six relief valves are staggered. Two valves are set at 1090 psig, two are set at 1095 psig, and two are set at 1100 psig. The operator will endeavor to place the set-point at these figures. However, the Allowable Value for each valve can be as much as ±24 psig. The as found value for at least 2 relief valves must be greater than the as found high reactor pressure scram value.

Six valves are provided for the automatic depressurization function, as described in 3.1.5. However, only five valves are required to prevent actuation of the safety valves, as discussed in the Technical Supplement to Petition to Increase Power Level, Section II.XV, letter, T. J. Brosnan to Peter A. Morris dated February 28, 1972, and letter, Philip D. Raymond to A. Giambusso, dated October 15, 1973.

~~The basis for the surveillance requirement is given in 4.1.5.~~

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.1 OXYGEN CONCENTRATION	4.3.1 OXYGEN CONCENTRATION
Applicability:	Applicability:
Applies to the limit on oxygen concentration within the primary containment system.	Applies to the periodic testing requirement for the primary containment system oxygen concentration.
Objective:	Objective:
To assure that in the event of a loss-of-coolant accident any hydrogen generation will not result in a combustible mixture within the primary containment system.	To assure that the oxygen concentration within the primary containment system is within required limits.
Specification:	Specification:
a. The primary containment atmosphere shall be reduced to less than four percent by volume oxygen concentration with nitrogen gas whenever the reactor coolant pressure is greater than 110 psig and the reactor is in the power operating condition, except as specified in "b" and "c" below.	~~At least once a week~~ oxygen concentration shall be determined. INSERT 1

The four percent by volume oxygen concentration eliminates the possibility of hydrogen combustion following a loss-of-coolant accident (Section VII-G.2.0 and Appendix E-II.5.2)*. The only way that significant quantities of hydrogen could be generated by metal-water reaction would be if the core spray system failed to sufficiently cool the core. As discussed in Section VII-A.2.0*, each core spray system will deliver, as a minimum, core spray sparger flow as shown on Figure VII-2*. In addition to hydrogen generated by metal-water reaction, significant quantities can be generated by radiolysis. (Technical Supplement to Petition for Conversion from Provisional Operating License to Full Term Operating License).

Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and deinerted as soon as possible in the plant shutdown. The probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a shutdown, is low enough that these "windows," when the primary containment is not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or deinerting.

If oxygen concentration is greater than or equal to four percent by volume at any time while in the power operating condition, with the exception of the relaxations allowed during startup and shutdown, oxygen concentration must be restored to less than four percent by volume within 24 hours. The 24 hour completion time is allowed when oxygen concentration is greater than or equal to four percent by volume because of the low probability and long duration of an event that would generate significant amounts of hydrogen occurring during this period.

If oxygen concentration cannot be restored to within limits within the required completion time, reactor coolant pressure must be reduced to less than or equal to 110 psig within 10 hours.

At reactor pressures of 110 psig or less, the reactor will have been shutdown for more than an hour and the decay heat will be at sufficiently low values so that fuel rods will be completely wetted by core spray. The fuel clad temperatures would not exceed the core spray water saturation temperature of about 344°F.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase the oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. ~~However, at least once a week, the oxygen concentration will be determined as added assurance that Specification 3.3.1 is being met.~~

INSERT 3

*FSAR

LIMITING CONDITION FOR OPERATION

3.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Applicability:

Applies to the interrelated parameters of pressure suppression system pressure and suppression chamber water temperature and level.

Objective:

To assure that the peak suppression chamber pressure does not exceed design values in the event of a loss-of-coolant accident.

Specification:

a. The downcomers in the suppression chamber shall have a minimum submergence of three and one half feet and a maximum submergence of four and one quarter feet whenever the reactor coolant system temperature is above 215°F and primary containment integrity is required.

b. During normal power operation, suppression chamber water temperature shall be less than or equal to 85°F.

SURVEILLANCE REQUIREMENT

4.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Applicability:

Applies to the periodic testing of the pressure suppression system pressure and suppression chamber water temperature and level.

Objective:

To assure that the pressure suppression system pressure and suppression chamber water temperature and level are within required limits.

Specification: INSERT 1

a. ~~At least once per day~~ the suppression chamber water level and temperature and pressure suppression system pressure shall be checked.

b. A visual inspection of the suppression chamber interior, including water line regions, shall be made ~~at each major refueling outage~~.

INSERT 1

BASES FOR 3.3.2 AND 4.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Continuous monitoring of suppression chamber water level and temperature and pressure suppression system pressure is provided in the control room. Alarms for these parameters are also provided in the control room.

To determine the status of the pressure suppression system, inspections of the suppression chamber interior surfaces ~~at each major refueling outage~~ with water at its normal elevation will be made. This will assure that gross defects are not developing.

INSERT 3

LIMITING CONDITION FOR OPERATION

3.3.4 PRIMARY CONTAINMENT ISOLATION VALVES

Applicability:

Applies to the operating status of the system of isolation valves on lines open to the free space of the primary containment.

Objective:

To assure that potential leakage paths from the primary containment in the event of a loss-of-coolant accident are minimized.

Specification:

a. Whenever the reactor coolant system temperature is greater than 215°F and primary containment integrity is required, all containment isolation valves on lines open to the free space of the primary containment shall be operable except as specified in 3.3.4b below.

b. In the event any isolation valve becomes inoperable the system shall be considered operable provided that within 4 hours at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

SURVEILLANCE REQUIREMENT

4.3.4 PRIMARY CONTAINMENT ISOLATION VALVES

Applicability:

Applies to the periodic testing requirements of the primary containment isolation valve system.

Objective:

To assure the operability of the primary containment isolation valves to limit potential leakage paths from the containment in the event of a loss-of-coolant accident.

Specification:

The primary containment isolation valves surveillance shall be performed as indicated below.

INSERT 1

a. ~~At least once per operating cycle~~ the operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and closure times.

INSERT 1

b. ~~At least once per quarter~~ all normally open power operated isolation valves shall be fully closed and reopened.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. If Specifications 3.3.4 a and b are not met, the reactor coolant system temperature shall be reduced to a value less than 215°F within ten hours.	c. ~~At least once per operating cycle,~~ each instrument-line flow check valve will be tested for operability. INSERT 1

The list of primary containment isolation valves is contained in the procedure governing controlled lists and has been removed from the Technical Specifications per Generic Letter 91-08. Revisions will be processed in accordance with Quality Assurance Program requirements.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Except where check valves are used as one or both of a set of double isolation valves, the isolation valves shall be capable of automatic initiation. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section VI-D[1]. For allowable leakage rate specification, see Section 3.3.3/4.3.3.

For the design basis loss-of-coolant accident fuel rod perforation would not occur until the fuel temperature reached 1700°F which occurs in approximately 100 seconds[2]. The required closing times for all primary containment isolation valves are established to prevent fission product release through lines connecting to the primary containment.

For reactor coolant system temperatures less than 215°F, the containment could not become pressurized due to a loss-of-coolant accident. The 215°F limit is based on preventing pressurization of the reactor building and rupture of the blowout panels.

~~The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1 x 10^-7^ that a line will not isolate (Fifth Supplement, p. 115)[3]. More frequent testing for valve operability results in a more reliable system.~~

In addition to routine surveillance as outlined in Section VI-D.1.0[1] each instrument-line flow check valve will be tested for operability. All instruments on a given line will be isolated at each instrument. The line will be purged by isolating the flow check valve, opening the bypass valves, and opening the drain valve to the equipment drain tank. When purging is sufficient to clear the line of non-condensibles and crud the flow-check valve will be cut into service and the bypass valve closed. The main valve will again be opened and the flow-check valve allowed to close. The flow-check valve will be reset by closing the drain valve and opening the bypass valve depressurizing part of the system. Instruments will be cut into service after closing the bypass valve. Repressurizing of the individual instruments assures that flow-check valves have reset to the open position. Alternatively, operability testing of excess flow check valves may be performed prior to installation using a test set-up that simulates an instrument line break condition.

INSERT 3

(1) UFSAR
(2) Nine Mile Point Nuclear Generation Station Unit 1 Safer/Corecool/GESTR-LOCA Loss of Coolant Accident Analysis, NEDC-31446P, Supplement 3, September, 1990.
(3) FSAR

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.6 VACUUM RELIEF

Applicability:

Applies to the operational status of the primary containment vacuum relief system.

Objective:

To assure the capability of the vacuum relief system in the event of a loss-of-coolant accident to:

a. Equalize pressures between the drywell and suppression chamber, and

b. Maintain containment pressure above the vacuum design values of the drywell and suppression chamber.

Specification:

a. When primary containment is required, all suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated above. Suppression chamber - drywell vacuum breakers shall be considered operable if:

(1) The valve is demonstrated to open fully with the applied force at all valve positions not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

4.3.6 VACUUM RELIEF

Applicability:

Applies to the periodic testing of the vacuum relief system.

Objective:

To assure the operability of the containment vacuum relief system to perform its intended functions.

Specification:

a. Periodic Operability Tests

INSERT 1

~~Once each month~~ and following any release of energy to the suppression chamber, each suppression chamber - drywell vacuum breaker shall be exercised. Operability of valves, position switches, and position indicators and alarms shall be verified monthly and following any maintenance on the valves and associated equipment.

LIMITING CONDITION FOR OPERATION

(2) The valve can be closed by gravity, when released after being opened by remote or manual means, to within not greater than the equivalent of 0.06 inch at the bottom of the disk.

(3) The position alarm system will annunciate in the control room if the valve opening exceeds the equivalent of 0.06 inch at the bottom of the disk.

b. Any drywell-suppression chamber vacuum breaker may be non-fully closed as indicated by the position indication and alarm systems provided that drywell to suppression chamber differential pressure decay rate is demonstrated to be not greater than 25% of the differential pressure decay rate for all vacuum breakers open the equivalent of 0.06 inch at the bottom of the disk.

SURVEILLANCE REQUIREMENT

b. Refueling Outage Tests

(1) All suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open.

(2) All suppression chamber - drywell vacuum breaker position indication and alarm systems shall be calibrated and functionally tested.

INSERT 1

(3) ~~Once each operating cycle,~~ each vacuum breaker valve shall be visually inspected to ensure proper maintenance and operation.

(4) A drywell to suppression chamber leak rate test shall demonstrate that with an initial differential pressure of not less than 1.0 psi, the differential pressure decay rate shall not exceed the equivalent of the leakage rate through a 1-inch orifice.

LIMITING CONDITION FOR OPERATION

c. When it is determined that one or more vacuum breaker valves are not fully closed as indicated by the position indication system at a time when such closure is required, the apparently malfunctioning vacuum breaker valve shall be exercised and pressure tested as specified in 3.3.6 b immediately and every 15 days thereafter until appropriate repairs have been completed.

d. One drywell-suppression chamber vacuum breaker may be secured in the closed position.

e. If Specifications 3.3.6 a, b, c, or d cannot be met, the situation shall be corrected within 24 hours or the reactor shall be placed in a cold shutdown condition within 24 hours.

f. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

(1) The three pressure suppression chamber reactor building vacuum breaker systems shall be operable at all times when the primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building air-operated vacuum breakers shall be ≤0.5 psid.

SURVEILLANCE REQUIREMENT

c. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

(1) The pressure suppression chamber-reactor building vacuum breaker systems and associated instrumentation, including set point, shall be checked for proper operation INSERT 1 ~~every three months.~~

(2) INSERT 1 ~~During each refueling outage,~~ each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.3.6.f(1) and each vacuum breaker shall be inspected and verified to meet design requirement.

Each drywell-suppression chamber vacuum breaker is equipped with two independent switches to indicate the opening of the valve disk. Redundant control room alarms are provided to permit detection of any drywell-suppression chamber vacuum breaker opening in excess of the described allowable limits. The containment design has been examined to establish the allowable bypass area between the drywell and suppression chamber as 0.053 square feet.

The limit on each individual valve will be set such that with all valves at their limit, the maximum value of cumulative leakage will not exceed the maximum allowable. The value will be at approximately 0.06 inch of disk travel off its seat and will be alarmed in the control room.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and between the suppression chamber and reactor building so that the structural integrity of the containment is maintained.

The vacuum relief system from the pressure suppression chamber to the reactor building consists of three vacuum relief breakers (3 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 1 psig; the external pressure is 2 psig.

The leak rate testing program is based on AEC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels.

Surveillance of the suppression chamber-reactor building vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. ~~Therefore, a testing frequency of three months for operability is considered justified for this equipment. Inspections and calibrations are performed during the refueling outages, this frequency is based on equipment quality, experience, and engineering judgment.~~ INSERT 1

~~During each refueling outage,~~ a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by approximately 1 psi with respect to the suppression pool pressure and then held constant. The subsequent suppression chamber transient will be monitored with a sufficiently sensitive pressure instrument. If the drywell pressure cannot be increased by 1 psi over the suppression chamber pressure, it would indicate existence of a significant leakage path which will be identified and eliminated before further drywell vacuum breaker testing.

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.7 CONTAINMENT SPRAY SYSTEM

Applicability:

Applies to the operating status of the containment spray system.

Objective:

To assure the capability of the containment spray system to limit containment pressure and temperature in the event of a loss-of-coolant accident.

Specification:

a. During all reactor operating conditions whenever reactor coolant temperature is greater than 215°F and fuel is in the reactor vessel and primary containment integrity is required; each of the two containment spray systems and the associated raw water cooling systems shall be operable except as specified in 3.3.7.b.

b. If a redundant component of a containment spray system becomes inoperable, Specification 3.3.7.a shall be considered fulfilled, provided that the component is returned to an operable condition within 15 days and that the additional surveillance required is performed.

4.3.7 CONTAINMENT SPRAY SYSTEM

Applicability:

Applies to the testing of the containment spray system.

Objective:

To verify the operability of the containment spray system.

Specification:

The containment spray system surveillance shall be performed as indicated below:

a. Containment Spray Pumps

INSERT 1

(1) ~~At least once per operating cycle,~~ automatic startup of the containment spray pump shall be demonstrated.

INSERT 1

(2) ~~At least once per quarter,~~ pump operability shall be checked.

b. Nozzles

Following maintenance that could result in nozzle blockage, a test shall be performed on the spray nozzles.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. If a redundant component in each of the containment spray systems or their associated raw water systems become inoperable, both systems shall be considered operable provided that the component is returned to an operable condition within 7 days and that the additional surveillance required is performed.

d. If a containment spray system or its associated raw water system becomes inoperable and all the components are operable in the other systems, the reactor may remain in operation for a period not to exceed 7 days.

e. If Specifications "a" or "b" are not met, shutdown shall begin within one hour and the reactor coolant shall be below 215°F within ten hours.

If both containment spray systems become inoperable the reactor shall be in the cold shutdown condition within ten hours and no work shall be performed on the reactor which could result in lowering the reactor water level to more than six feet, three inches (-10 inches indicator scale) below minimum normal water level (Elevation 302'9").

c. Raw Water Cooling Pumps

INSERT 1 ~~At least once per quarter~~ manual startup and operability of the raw water cooling pumps shall be demonstrated.

d. Surveillance with Inoperable Components

When a component or system becomes inoperable its redundant component or system shall be verified to be operable immediately and ~~daily~~ thereafter.

INSERT 1

LIMITING CONDITION FOR OPERATION

f. The containment spray system shall be considered operable by verifying that lake water temperature does not exceed 83°F.

g. If specification "f" cannot be met commence shutdown within one hour and be in hot shutdown within 8 hours and cold shutdown within 24 hours.

SURVEILLANCE REQUIREMENT

f. Lake Water Temperature

INSERT 1

Record ~~at least once per 24 hours~~, and at least once per 8 hours when latest recorded water temperature is greater than or equal to 75°F and at least once per 4 hours when the latest recorded water temperature is greater than or equal to 79°F.

BASES FOR 3.3.7 AND 4.3.7 CONTAINMENT SPRAY SYSTEM

In conjunction with containment spray pump operation during each operating cycle, the raw water pumps and associated cooling system performance will be observed. The containment spray system shall be capable of automatic initiation from simultaneous low-low reactor water level and high containment pressure. The associated raw water cooling system shall be capable of manual actuation. Operation of the containment spray system involves spraying water into the atmosphere of the containment. Therefore, periodic system tests are not practical. Instead separate testing of automatic containment spray pump startup will be performed during each operating cycle. During pump operation, water will be recycled to the suppression chamber. Also, tests to verify that the drywell and torus spray nozzles are free from obstructions will be performed following maintenance that could result in nozzle blockage. As an alternative, a visual inspection (e.g., boroscope) of the nozzles or piping could be utilized in lieu of an air test if a visual inspection is determined to provide an equivalent or more effective post-maintenance test. A visual inspection may be more effective if the potential for material intrusion is localized and the affected area is accessible. Maintenance that could result in nozzle blockage would be those maintenance activities on any loop of the containment spray system where the Foreign Material Exclusion program controls were deemed ineffective. For activities such as valve repair/replacement, a visual inspection would be the preferred post-maintenance test since small debris in a localized area is the most likely concern. An air test may be appropriate following an event where a large amount of debris potentially entered the system or water was actually discharged through the spray nozzles. Design features are discussed in Volume I, Section VII-B.2.0 (page VII-19)*. The valves in the containment spray system are normally open and are not required to operate when the system is called upon to operate.

~~The test interval between operating cycle results in a system failure probability of 1.1 x 10~~$\cancel{^{-6}}$ ~~(Fifth Supplement, page 115)* and is consistent with practical considerations. Pump operability will be demonstrated on a more frequent basis and will provide a more reliable system.~~

INSERT 3

*FSAR

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.4.1 LEAKAGE RATE

Applicability:

Applies to the leakage rate of the secondary containment.

Objective:

To specify the requirements necessary to limit exfiltration of fission products released to the secondary containment as a result of an accident.

Specification:

At all times when secondary containment integrity is required, the reactor building leakage rate as determined by Specification 4.4.1 shall not exceed 1600 cfm. If this cannot be met after a routine surveillance check, then the actions listed below shall be taken:

a. Suspend any of the following activities:

1. Handling of recently irradiated fuel in the reactor building.
2. Irradiated fuel cask operations in the reactor building.
3. Operations with a potential for draining the reactor vessel (OPDRVs).

b. Restore the reactor building leakage rates to within specified limits within 4 hours or initiate normal orderly shutdown and be in a cold shutdown condition within 10 hours.

4.4.1 LEAKAGE RATE

Applicability:

Applies to the periodic testing requirements of the secondary containment leakage rate.

Objective:

To assure the capability of the secondary containment to maintain leakage within allowable limits.

Specification: INSERT 1

~~Once during each operating cycle~~ isolate the reactor building and start emergency ventilation system fan to demonstrate negative pressure in the building relative to external static pressure. The fan flow rate shall be varied so that the building internal differential pressure is at least as negative as that on Figure 3.4.1 for the wind speed at which the test is conducted. The fan flow rate represents the reactor building leakage referenced to zero mph with building internal pressure at least 0.25 inch of water less than atmospheric pressure. The test shall be done at wind speeds less than 20 miles per hour.

If the wind direction was not from the direction which gave the least reactor building leakage, building internal pressure would not be as negative as Figure 3.4.1 indicates. Therefore, to reduce pressure, the fan flow rate would have to be increased. This erroneously indicates that reactor building leakage is greater than if wind direction were accounted for. If wind direction were accounted for, another pressure curve could be used which was less negative. This would mean that less fan flow (or measured leakage) would be required to establish building pressure. However, for simplicity it is assumed that the test is conducted during conditions leading to the least leakage while the accident is assumed to occur during conditions leading to the greatest reactor building leakage.

As discussed in the Second Supplement and Fifth Supplement, the pressure for Figure 3.4.1 is independent of the reactor building leakage rate referenced to zero mph wind speed at a negative differential pressure of 0.25 inch of water. Regardless of the leakage rate at these design conditions, the pressure versus wind speed relationship remains unchanged for any given wind direction.

By requiring the reactor building pressure to remain within the limits presented in Figure 3.4.1 and a reactor building leakage rate of less than 1600 cfm, exfiltration would be prevented. This would assure that the leakage from the primary containment is directed through the filter system and discharged from the 350-foot stack.

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.4.2 REACTOR BUILDING INTEGRITY - ISOLATION VALVES	4.4.2 REACTOR BUILDING INTEGRITY - ISOLATION VALVES

3.4.2 REACTOR BUILDING INTEGRITY - ISOLATION VALVES

Applicability:

Applies to the operational status of the reactor building isolation valves.

Objective:

To assure that fission products released to the secondary containment are discharged to the environment in a controlled manner using the emergency ventilation system.

Specification:

a. The normal Ventilation System isolation valves shall be operable at all times when secondary containment integrity is required.

b. If Specification 3.4.2.a is not met, then the actions listed below shall be taken:

1. The reactor shall be in the cold shutdown condition within ten hours.

2. Suspend any of the following activities:

   a. Handling of recently irradiated fuel in the reactor building,

   b. Irradiated fuel cask handling operations in the reactor building,

   c. Operations with a potential for draining the reactor vessel (OPDRVs).

4.4.2 REACTOR BUILDING INTEGRITY - ISOLATION VALVES

Applicability:

Applies to the periodic testing requirements of the reactor building isolation valves.

Objective:

To assure the operability of the reactor building isolation valves.

Specification:

INSERT 1 ~~At least once per operating cycle~~, automatic initiation of valves shall be checked.

Isolation of the reactor building occurs automatically upon high radiation of the normal building exhaust ducts or from high radiation at the refueling platform (See 3.6.2). Isolation will assure that any fission products entering the reactor building will be routed to the emergency ventilation system prior to discharge to the environment (Section VII-H.3.0 of the FSAR).

The two principal accidents for which the reactor building isolation valves must be operable are a loss of coolant accident (LOCA) and a refueling accident involving "recently irradiated" fuel. In addition to these limiting events, events occurring during handling of an irradiated fuel cask and operations with a potential for draining the reactor vessel (OPDRVs) can be postulated to cause a fission product release. During these events, the reactor building would be the only barrier to a release to the environment. Thus, the reactor building isolation valves are required to be operable during handling of an irradiated fuel cask and during OPDRVs.

The Refueling Accident analysis is based on an alternative source term (AST) methodology (10 CFR 50.67 and Regulatory Guide 1.183). This analysis concluded that the calculated total effective dose equivalent (TEDE) values to the control room occupants, the exclusion area boundary, and the low population zone are well below the TEDE criteria established in 10 CFR 50.67 without crediting reactor building integrity or operation of the reactor building emergency ventilation system (RBEVS), as long as the fuel is allowed to decay for at least 24 hours following reactor shutdown. As a result, "recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within 24 hours; i.e., reactor fuel that has decayed less than 24 hours following reactor shutdown. Therefore, reactor building integrity is not required and the reactor building isolation valves are not required to be operable during movement of decayed irradiated fuel that is no longer considered "recently irradiated." Conversely, reactor building integrity is required and the reactor building isolation valves are required to be operable during movement of recently irradiated fuel assemblies.

INSERT 3

LIMITING CONDITION FOR OPERATION

3.4.3 ACCESS CONTROL

Applicability:

Applies to the access control to the reactor building.

Objective:

To specify the requirements necessary to assure the integrity of the secondary containment system.

Specification:

a. At all times when secondary containment integrity is required, the following conditions will be met:

   1. Only one door in each of the double-doored access ways shall be opened at one time.

   2. Only one door or closeup of the railroad bay shall be opened at one time.

   3. The core spray and containment spray pump compartments' doors shall be closed at all times except during passage in order to consider the core spray system and the containment spray system operable.

SURVEILLANCE REQUIREMENT

4.4.3 ACCESS CONTROL

Applicability:

Applies to the periodic checking of the condition of portions of the reactor building.

Objective:

To assure that pump compartments are properly closed at all times and to assure the integrity of the secondary containment system by verifying that reactor building access doors are closed, as required by Specifications 3.4.3.a.1 and 3.4.3.a.2.

Specification:

a. The core and containment spray pump compartments shall be checked ~~once per week~~ and after each entry.

INSERT 1

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. If these conditions cannot be met, then the actions listed below shall be taken:	b. Verify ~~at least once per 31 days~~ that: INSERT 1
1. If in the power operating condition, restore reactor building integrity within 4 hours or be in at least the hot shutdown condition within the next 12 hours and in the cold shutdown condition within the following 24 hours.	1. At least one door in each access to the secondary containment is closed.
OR	2. At least one door or closeup of the railroad bay is closed.
If the reactor coolant system temperature is above 215°F, restore reactor building integrity within 4 hours or be in cold shutdown within the following 24 hours.	
2. Suspend any of the following activities:	
a. Handling of recently irradiated fuel in the reactor building,	
b. Irradiated fuel cask handling operations in the reactor building,	
c. Operations with a potential for draining the reactor vessel (OPDRVs).	

The secondary containment is designed to minimize any ground level release of radioactive materials that might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service. The reactor building provides primary containment during periods when the reactor is shutdown, the drywell is open, and activities are ongoing that require secondary containment to be in effect.

There are two principal accidents for which credit is taken for reactor building (secondary containment) integrity. These are a loss of coolant accident (LOCA) and a refueling accident involving "recently irradiated" fuel. The reactor building performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the reactor building structure will be treated by the Reactor Building Emergency Ventilation System (RBEVS) prior to discharge to the environment.

In addition to these limiting events, events occurring during handling of an irradiated fuel cask and operations with a potential for draining the reactor vessel (OPDRVs) can be postulated to cause a fission product release. During these events, the reactor building would be the only barrier to a release to the environment. Thus, reactor building integrity is required during handling of an irradiated fuel cask and during OPDRVs.

The Refueling Accident analysis is based on an alternative source term (AST) methodology (10 CFR 50.67 and Regulatory Guide 1.183). This analysis concluded that the calculated total effective dose equivalent (TEDE) values to the control room occupants, the exclusion area boundary, and the low population zone are well below the TEDE criteria established in 10 CFR 50.67 without crediting reactor building integrity, operation of the RBEVS, or operation of the Control Room Air Treatment System (CRATS), as long as the fuel is allowed to decay for at least 24 hours following reactor shutdown. As a result, "recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within 24 hours; i.e., reactor fuel that has decayed less than 24 hours following reactor shutdown. Therefore, reactor building integrity is not required during movement of decayed irradiated fuel that is no longer considered "recently irradiated." Conversely, reactor building integrity is required during movement of recently irradiated fuel assemblies.

As discussed in Section VI-F* all access openings of the reactor building have as a minimum two doors in series. Appropriate local alarms and control room indicators are provided to always insure that reactor building integrity is maintained. Surveillance of the reactor building access doors provides additional assurance that reactor building integrity is maintained.

Maintaining closed doors on the pump compartments ensures that suction to the core and containment spray pumps is not lost in case of a gross leak from the suppression chamber.

INSERT 3

*FSAR

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.4.4 EMERGENCY VENTILATION SYSTEM

Applicability:

Applies to the operating status of the emergency ventilation system.

Objective:

To assure the capability of the emergency ventilation system to minimize the release of radioactivity to the environment in the event of an incident within the primary containment or reactor building.

Specification:

a. Except as specified in Specification 3.4.4e below, both circuits of the emergency ventilation system shall be operable at all times when secondary containment integrity is required.

b. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal when tested in accordance with ANSI N.510-1980.

4.4.4 EMERGENCY VENTILATION SYSTEM

Applicability:

Applies to the testing of the emergency ventilation system.

Objective:

To assure the operability of the emergency ventilation system.

Specification:

Emergency ventilation system surveillance shall be performed as indicated below:

INSERT 1

a. ~~At least once per operating cycle, not to exceed 24 months~~, the following conditions shall be demonstrated:

(1) Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at the system rated flow rate ($\pm$ 10%).

(2) Deleted

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

c. The results of laboratory carbon sample analysis shall show ≥95% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 at 30°C and 95% R.H.

d. *Fans shall be shown to operate within ±10%* design flow.

e. During reactor operation, including when the reactor coolant system temperature is above 215°F, from and after the date that one circuit of the emergency ventilation system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other emergency ventilation circuit shall be operable.

During handling of recently irradiated fuel in the reactor building, handling of an irradiated fuel cask in the reactor building, and operations with a potential for draining the reactor vessel (OPDRVs), from and after the date that one circuit of the emergency ventilation system is made or found to be inoperable for any reason, recently irradiated fuel handling in the reactor building, irradiated fuel cask handling in the reactor building, or OPDRVs are permissible during the succeeding seven days unless such circuit is sooner made operable, provided that

b. The tests and sample analysis of Specification 3.4.4b, c and d shall be performed ~~at least once per operating cycle or once every 24 months~~, or after 720 hours of system operation, whichever occurs first or following significant painting, fire *or chemical release in any ventilation zone* communicating with the system. INSERT 1

c. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

d. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

e. Each circuit shall be operated at least 15 minutes ~~every month.~~ INSERT 1

f. Test sealing of gaskets for housing doors downstream of the HEPA filters and charcoal adsorbers shall be performed at and in conformance with each test performed for compliance with Specification 4.4.4b and Specification 3.4.4b.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
during such seven days all active components of the other emergency ventilation circuit shall be operable. Recently irradiated fuel handling in the reactor building, irradiated fuel cask handling in the reactor building, or OPDRVs may continue beyond seven days provided the operable emergency ventilation circuit is in operation.	g. ~~At least once per operating cycle, not to exceed 24 months~~ INSERT 1, automatic initiation of each branch of the emergency ventilation system shall be demonstrated.
f. If these conditions cannot be met, within 36 hours, the reactor shall be placed in a condition for which the emergency ventilation system is not required.	h. ~~At least once per operating cycle, not to exceed 24 months~~ INSERT 1, manual operability of the bypass valve for filter cooling shall be demonstrated.
	i. When one circuit of the emergency ventilation system becomes inoperable all active components in the other emergency ventilation circuit shall be verified to be operable within two hours and ~~daily~~ INSERT 1 thereafter.

The emergency ventilation system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. Both emergency ventilation system fans are designed to automatically start upon high radiation in the reactor building ventilation duct or at the refueling platform and to maintain the reactor building pressure to the design negative pressure so as to minimize in-leakage. Should one system fail to start, the redundant system is designed to start automatically. Each of the two fans has 100 percent capacity.

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 95 percent, which is derived from applying a safety factor of 2 to the charcoal filter efficiency of 90 percent assumed in analysis of design basis accidents. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 50.67 acceptance criteria for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Only one of the two emergency ventilation systems is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling activities may continue while repairs are being made. If neither circuit is operable, the plant is brought to a condition where the emergency ventilation system is not required.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test should allow for charcoal sampling to be conducted using an ASTM D3803-1989 approved method. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent meeting the physical property specifications of Table 5-1 of ANSI 509-1980.

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the operating status of the control room air treatment system and Control Room Envelope (CRE) boundary.

---------------------------NOTE---------------------------
The CRE boundary may be opened intermittently under administrative control.

Objective:

To assure the capability of the control room air treatment system to minimize the amount of radioactivity or other gases entering the control room in the event of an incident.

Specification:

a. Except as specified below, the control room air treatment system shall be operable for the following conditions:

   1. Power operating condition, and whenever the reactor coolant system temperature is greater than 212°F,

   2. Whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building, and

   3. During operations with a potential for draining the reactor vessel (OPDRVs).

b. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal when tested in accordance with ANSI N.510-1980.

4.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

Applicability:

Applies to the testing of the control room air treatment system and CRE boundary.

Objective:

To assure the operability of the control room air treatment system.

Specification: INSERT 1

a. ~~At least once per operating cycle, or once every 24 months, whichever occurs first,~~ the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 1.5 inches of water at system design flow rate (± 10%).

INSERT 1

b. The tests and sample analysis of Specification 3.4.5b, c and d shall be performed ~~at least once per operating cycle or once every 24 months~~, or after 720 hours of system operation, whichever occurs first or following significant painting, fire or chemical release in any ventilation zone communicating with the system.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. The results of laboratory carbon sample analysis shall show ≥95% radioactive methyl iodine removal when tested in accordance with ASTM D3803-1989 at 30°C and 95% R.H.	c. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
d. Fans shall be shown to operate within ±10% design flow.	d. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal absorber bank or after any structural maintenance on the system housing.
e. From and after the date that the control room air treatment system is made or found to be inoperable for any reason, except for an inoperable CRE boundary during the power operating condition, restore the system to operable within the succeeding seven days.	e. The system shall be operated at least 15 minutes ~~every month.~~ INSERT 1
f. If the control room air treatment system is made or found to be inoperable due to an inoperable CRE boundary during the power operating condition: immediately initiate action to implement mitigating actions; within 24 hours, verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits; and within 90 days, restore the CRE boundary to operable status.	f. INSERT 1 ~~At least once per operating cycle, not to exceed 24 months~~, automatic initiation of the control room air treatment system shall be demonstrated.
g. If Specifications 3.4.5.e or 3.4.5.f cannot be met during the power operating condition, or when reactor coolant system temperature is greater than 212°F, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 36 hours.	g. In accordance with the frequency and specifications of the Control Room Envelope Habitability Program, perform required CRE unfiltered air inleakage testing.
h. If Specification 3.4.5.e cannot be met whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building, or during OPDRVs: immediately suspend handling of recently irradiated fuel or the irradiated fuel cask in the reactor building; and immediately initiate action to suspend OPDRVs.	

(Ref. 4). Options for restoring the CRE boundary to operable status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to operable status.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorber. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 95 percent, which is derived from applying a safety factor of 2 to the charcoal filter efficiency of 90 percent assumed in analyses of design basis accidents. If the efficiencies of the HEPA filter and charcoal adsorbers are as specified, adequate radiation protection will be provided such that resulting doses will be less than the allowable levels stated in 10 CFR 50.67. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 1.5 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. ~~Pressure drop should be determined at least once per operating cycle to show system performance capability.~~

The frequency of tests and sample analysis are necessary to show the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test should allow for charcoal sampling to be conducted using an ASTM D3803-1989 approved method. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent meeting the physical property specifications of Table 5-1 of ANSI 509-1980. The replacement charcoal for the adsorber tray removed for the test should meet the same adsorbent quality. Any HEPA filters found defective shall be replaced with filters qualified pursuant to ANSI 509-1980.

INSERT 3

The two principal accidents for which the Control Room Air Treatment System (CRATS) must be operable are a loss of coolant accident (LOCA) and a refueling accident involving "recently irradiated" fuel. In addition to these limiting events, events occurring during handling of an irradiated fuel cask and operations with a potential for draining the reactor vessel (OPDRVs) can be postulated to cause a fission product release. Thus, the CRATS is required to be operable during handling of an irradiated fuel cask and during OPDRVs.

The Refueling Accident analysis is based on an alternative source term (AST) methodology (10 CFR 50.67 and Regulatory Guide 1.183). This analysis concluded that the calculated total effective dose equivalent (TEDE) values to the control room occupants, the exclusion area boundary, and the low population zone are well below the TEDE criteria established in 10 CFR 50.67 without crediting operation of the CRATS, as long as the fuel is allowed to decay for at least 24 hours following reactor shutdown. As a result, "recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within 24 hours; i.e., reactor fuel that has decayed less than 24 hours following reactor shutdown. Therefore, the CRATS is not required to be operable during movement of decayed irradiated fuel that is no longer considered "recently irradiated." Conversely, the CRATS is required to be operable during movement of recently irradiated fuel assemblies.

Operation of the system for 15 minutes ~~every month~~ will demonstrate operability of the filters and adsorber system.

INSERT 1

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

INSERT 3

References:

1. UFSAR, Section III.B.

2. Regulatory Guide 1.196, Revision 0, May 2001.

3. NEI 99-03, "Control Room Habitability Assessment." June 2001.

4. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of the Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability. " (ADAMS Accession No. ML040160868).

LIMITING CONDITION FOR OPERATION

3.5.1 SOURCE RANGE MONITORS

Applicability:

Applies to the operating status of the source range monitors.

Objective:

To assure the capability of the source range monitors to provide neutron flux indication required for reactor shutdown and startup and refueling operations.

Specification:

Whenever the reactor is in the shutdown, refueling or power operating conditions (unless the IRM's or APRM's are on scale) or whenever core alterations are being made at least three SRM channels will be operable except as noted in Specification 3.5.3. To be considered operable, the following conditions must be satisfied:

a. Inserted to normal operating level and available for monitoring the core. May be withdrawn as long as a minimum count rate of 100 cps is maintained.

SURVEILLANCE REQUIREMENT

4.5.1 SOURCE RANGE MONITORS

Applicability:

Applies to the periodic testing of the source range monitors.

Objective:

To assure the operability of the source range monitors to monitor low-level neutron flux.

Specification:

The source range monitoring system surveillance will be performed as indicated below.

~~During each operating cycle~~ - check in-core to out-of-core signal ratio and minimum count rate.

INSERT 1

BASES FOR 3.5.1 AND 4.5.1 SOURCE RANGE MONITORS

The SRM's are provided to monitor the core during periods of Station shutdown and to guide the operator during refueling operations and Station startup. Requiring three operative SRM's will ensure adequate coverage for all possible critical configurations produced by fuel loading or dispersed withdrawals of control rods during Station startup. Allowing withdrawal of the SRM while maintaining a high count rate will extend the operating range of the SRM's. Evaluation of the SRM operation is presented in Section VIII-C.1.2.1 of the FSAR.

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.2 REFUELING PLATFORM INTERLOCK	4.5.2 REFUELING PLATFORM INTERLOCK

3.5.2 REFUELING PLATFORM INTERLOCK

Applicability:

Applies to the refueling platform on interlocks.

Objective:

To assure that a loaded refueling platform hoist is never over the core when one or more control rods are withdrawn.

Specification:

During the refueling condition with the mode switch in the "refuel" position the following interlocks must be operative:

a. Control rod withdrawal block with a fuel assembly on the hoist over the reactor core.

b. With a control rod withdrawn from the core the refuel platform, if loaded with a fuel assembly, is blocked from travelling over the core.

c. If the interlocks for either "a" or "b" or both are not operable, double procedural control will be used to ensure that "a" and "b" are met.

4.5.2 REFUELING PLATFORM INTERLOCK

Applicability:

Applies to the periodic testing requirements for the refueling platform interlocks.

Objective:

To assure the operability of the refueling platform interlock.

Specification:

The refueling platform interlocks shall be tested prior to any fuel handling with the head off the reactor vessel, ~~at weekly intervals~~ thereafter until no longer required and following any repair work associated with the interlocks.

and in accordance with the Surveillance Frequency Control Program

BASES FOR 3.5.2 AND 4.5.2 REFUELING PLATFORM INTERLOCK

The control rod withdrawal block and refueling platform travel blocks are provided to back up normal procedural controls to prevent inadvertent large reactivity additions to the core. These interlocks are provided even though no more than one control rod can be removed from the core at a time during refueling with the mode switch in the "refuel" position. Even in the fresh fully loaded core if a new assembly is dropped into a vacant position adjacent to the withdrawn rod, no excursion would result. This is discussed in detail in Appendix E-II.3.0 of the FSAR.

There are normally two Station personnel directly involved in refueling the reactor, one in the control room and one at the platform. If the interlocks are inoperable, an additional person will check that "a" and "b" are not violated.

$\boxed{\text{INSERT 3}}$

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.1 MECHANICAL VACUUM PUMP ISOLATION	4.6.1 MECHANICAL VACUUM PUMP ISOLATION
a. (Deleted)	a. (Deleted) INSERT 1
b. The mechanical vacuum pump line shall be capable of automatic isolation by closure of the air-operated valve upstream of the pumps. The signal to initiate isolation shall be from high radioactivity (five times normal) in the main steam line.	b. ~~At least once during each operating cycle~~ (prior to startup), verify automatic securing and isolation of the mechanical vacuum pump.

BASES FOR 3.6.1b AND 4.6.1b MECHANICAL VACUUM PUMP ISOLATION

The purpose of isolating the mechanical vacuum pump line is to limit release of activity from the main condenser during a control rod drop accident. During the accident, fission products would be transported from the reactor through the main-steam lines to the main condenser. The fission product radioactivity would be sensed by the main-steam line radioactivity monitors and initiate isolation.

INSERT 3

LIMITING CONDITION FOR OPERATION

3.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the operability of the plant instrumentation that performs a safety function.

Objective:

To assure the operability of the instrumentation required for safe operation.

Specification:

a. The set points, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Tables 3.6.2a to 3.6.2l.

   If the requirements of a table are not met, the actions listed below for the respective type of instrumentation shall be taken.

   (1) Instrumentation that initiates scram - control rods shall be inserted, unless there is no fuel in the reactor vessel.

SURVEILLANCE REQUIREMENT

4.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the surveillance of the instrumentation that performs a safety function.

Objective:

To verify the operability of protective instrumentation.

Specification:

a. Sensors and instrument channels shall be checked, tested and calibrated ~~at least as frequently as listed in Tables 4.6.2a to 4.6.2l.~~

at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Tables 4.6.2a to 4.6.2l

LIMITING CONDITION FOR OPERATION

b. During operation with the Core Maximum Fraction of Limiting Power Density (CMFLPD) greater than the Fraction of Rated Thermal Power (FRTP), either:

(1) The APRM scram and rod block settings shall be reduced to the values given by the equations in Specification 2.1.2a; or

(2) The APRM gain shall be adjusted in accordance with Specification 2.1.2a; or

(3) The power distribution shall be changed such that the CMFLPD no longer exceeds FRTP.

SURVEILLANCE REQUIREMENT

c. During reactor power operation at ≥25 percent rated thermal power, the Core Maximum Fraction of Limiting Power Density (CMFLPD) shall be checked ~~daily~~ and the flow-referenced APRM scram and rod block signals shall be adjusted, if necessary, as specified by Specification 2.1.2a.

INSERT 1

TABLE 4.6.2a

INSTRUMENTATION THAT INITIATES SCRAM

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Manual Scram	None	~~Once per week~~ Note 1	Note 1
(2)	High Reactor Pressure	None	~~Once per 3 months~~[1] Note 1	~~Once per 3 months~~[1] Note 1
(3)	High Drywell Pressure	None	~~Once per 3 months~~[1] Note 1	~~Once per 3 months~~[1] Note 1
(4)	Low Reactor Water Level	~~Once/day~~ Note 1	~~Once per 3 months~~[1] Note 1	~~Once per 3 months~~[1] Note 1
(5)	High Water Level Scram Discharge Volume	None	~~Once per 3 months~~ Note 1	~~Once per 3 months~~ Note 1
(6)	Main-Steam-Line Isolation Valve Position	None	~~Once per 3 months~~ Note 1	~~Once per operating cycle~~ Note 1
(7)	Deleted			

TABLE 4.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(8) Shutdown Position of Reactor Mode Switch	None	Once during each major refueling outage	None
(9) Neutron Flux			
(a) IRM			
(i) Upscale	Note 1 ~~Once per shift~~(f)	Note 1 ~~Once per week~~(g)	Note 1 ~~Once per operating cycle~~(n)
(ii) Inoperative	Note 1 ~~Once per shift~~(f)	Note 1 ~~Once per week~~(g)	Note 1 ~~Once per operating cycle~~(n)
(b) APRM			
(i) Upscale	None	Note 1 ~~Once per 3 months~~	Note 1 ~~Once per week~~(m) Note 1 ~~Once per 3 months~~(n)
(ii) Inoperative	None	Note 1 ~~Once per 3 months~~	None
(10) Turbine Stop Valve Closure	None	Note 1 ~~Once per 3 months~~	Note 1 ~~Once per operating cycle~~
(11) Generator Load Rejection	None	Note 1 ~~Once per 3 months~~	Note 1 ~~Once per 3 months~~

(a) May be bypassed when necessary for containment inerting.

(b) May be bypassed in the refuel and shutdown positions of the reactor mode switch with a keylock switch.

(c) May be bypassed in the refuel and startup positions of the reactor mode switch when reactor pressure is less than 600 psi, or for the purpose of performing reactor coolant system pressure testing and/or control rod scram time testing with the reactor mode switch in the refuel position.

(d) No more than one of the four IRM inputs to each trip system shall be bypassed.

(e) No more than two C or D level LPRM inputs to an APRM shall be bypassed and only four LPRM inputs to an APRM shall be bypassed in order for the APRM to be considered operable. No more than one of the four APRM inputs to each trip system shall be bypassed provided that the APRM in the other instrument channel in the same core quadrant is not bypassed. A Traversing In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing.

(f) Verify SRM/IRM channels overlap during startup after the mode switch has been placed in startup. Verify IRM/APRM channels overlap at least 1/2 decade during entry into startup from run (normal shutdown) if not performed within the previous 7 days.

(g) Within 24 hours before startup, if not performed within the previous 7 days. Not required to be performed during shutdown until 12 hours after entering startup from run.

(h) Each of the four isolation valves has two limit switches. Each limit switch provides input to one of two instrument channels in a single trip system.

(i) May be bypassed when reactor power level is below 45%.

(j) Trip upon loss of oil pressure to the acceleration relay.

(k) May be bypassed when placing the reactor mode switch in the SHUTDOWN position and all control rods are fully inserted.

(l) ~~Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2a, the primary sensor will be calibrated and tested once per operating cycle.~~ INSERT 3

(m) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during reactor operation when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the difference is greater than +2.0/-1.9% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 2.1.2a shall not be included in determining the difference.

(n) Neutron detectors are excluded.

(n) Deleted.

(o) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same trip system is monitoring that parameter.

With one channel required by Table 3.6.2a inoperable in one or more Parameters, place the inoperable channel and/or that trip system in the tripped condition* within 12 hours.

With two or more channels required by Table 3.6.2a inoperable in one or more Parameters:

1. Within one hour, verify sufficient channels remain Operable or tripped* to maintain trip capability for the Parameter, and

2. Within 6 hours, place the inoperable channel(s) in one trip system and/or that trip system** in the tripped condition*, and

3. Within 12 hours, restore the inoperable channels in the other trip system to an Operable status or tripped*.

Otherwise, take the ACTION required by Specification 3.6.2a for that Parameter.

* An inoperable channel or trip system need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to Operable status within the required time, the ACTION required by Specification 3.6.2a for the parameter shall be taken.

** This ACTION applies to that trip system with the most inoperable channels; if both trip systems have the same number of inoperable channels, the ACTION can be applied to either trip system.

(p) May be bypassed during reactor coolant system pressure testing and/or control rod scram time testing.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2a

TABLE 4.6.2b

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

Parameter	**Sensor Check**	**Instrument Channel Test**	**Instrument Channel Calibration**
PRIMARY COOLANT ISOLATION (Main Steam, Cleanup and Shutdown Cooling)			
(1) Low-Low Reactor Water Level	Note 1 ~~Once/day~~	Note 1 ~~Once per 3 months~~(d)	Note 1 ~~Once per 3 months~~(d)
(2) Manual	---	Note 1 ~~Once during each major refueling outage~~	---
MAIN-STEAM-LINE ISOLATION			
(3) High Steam Flow Main-Steam Line	Note 1 ~~Once/day~~	Note 1 ~~Once per 3 months~~(d)	Note 1 ~~Once per 3 months~~(d)
(4) Deleted			
(5) Low Reactor Pressure	Note 1 ~~Once/day~~	Note 1 ~~Once per 3 months~~(d)	Note 1 ~~Once per 3 months~~(d)

TABLE 4.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(6) Low-Low-Low Condenser Vacuum	None	Note 1 ~~Once during each major refueling outage~~	Note 1 ~~Once during each major refueling outage~~
(7) High Temperature Main-Steam-Line Tunnel	None	Note 1 ~~Once during each major refueling outage~~	Note 1 ~~Once during each major refueling outage~~
CLEANUP SYSTEM ISOLATION			
(8) High Area Temperature	Note 1 ~~Once/week~~	Note 1 ~~Once during each major refueling outage~~	Note 1 ~~Once during each major refueling outage~~
SHUTDOWN COOLING SYSTEM ISOLATION			
(9) High Area Temperature	Note 1 ~~Once/week~~	Note 1 ~~Once per operating cycle~~	Note 1 ~~Once per operating cycle~~

TABLE 4.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
CONTAINMENT ISOLATION			
(10) Low-Low Reactor Water Level	Note 1 ~~Once/day~~	Note 1 ~~Once per 3 months~~(d)	Note 1 ~~Once per 3 months~~(d)
(11) High Drywell Pressure	Note 1 ~~Once/day~~	Note 1 ~~Once per 3 months~~(d)	Note 1 ~~Once per 3 months~~(d)
(12) Manual	---	Note 1 ~~Once during each operating cycle~~	---

NOTES FOR TABLES 3.6.2b and 4.6.2b

(a) May be bypassed in the refuel and startup positions of the reactor mode switch when reactor pressure is less than 600 psi.

(b) May be bypassed when necessary for containment inerting.

(c) May be bypassed in the shutdown mode whenever the reactor coolant system temperature is less than 215°F.

(d) ~~Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2b, the primary sensor will be calibrated and tested once per operating cycle.~~ INSERT 3

(e) Deleted.

(f) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that Parameter.

With the number of Operable Channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within

   a. 12 hours for Parameters common to SCRAM Instrumentation, and
   b. 24 hours for Parameters not common to SCRAM Instrumentation.

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

With the number of Operable Channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour.

and

2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within

      (1) 12 hours for Parameters common to SCRAM Instrumentation, and
      (2) 24 hours for Parameters not common to SCRAM Instrumentation.

   or

   b. take the ACTION required by Specification 3.6.2a for that Parameter.

(g) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that Parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels for the Operable Trip System, either

1. Place the inoperable channel(s) in the tripped condition within 24 hours.

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

(h) Only applicable during startup mode while operating in IRM range 10.

(i) May be bypassed in the cold shutdown condition.

(j) In the cold shutdown and refueling conditions, only one Operable Trip System is required provided shutdown cooling system integrity is maintained. With one of the two required Operable Channels in the required Trip System not operable, place the inoperable channel in the tripped condition within 12 hours. Otherwise, either:

1. Immediately initiate action to restore the channel to operable status.

or

2. Immediately initiate action to isolate the shutdown cooling system.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2b.

TABLE 4.6.2c

INSTRUMENTATION THAT INITIATES OR ISOLATES EMERGENCY COOLING

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
EMERGENCY COOLING INITIATION			
(1) High Reactor Pressure	None	Note 1 ~~Once per 3 months~~(c)	Note 1 ~~Once per 3 months~~(c)
(2) Low-Low Reactor Water Level	Note 1 ~~Once/day~~	Note 1 ~~Once per 3 months~~(c)	Note 1 ~~Once per 3 months~~(c)
EMERGENCY COOLING ISOLATION (for each of two systems)			
(3) High Steam Flow Emergency Cooling System	None	Note 1 ~~Once per 3 months~~(c)	Note 1 ~~Once per 3 months~~(c)

NOTES FOR TABLES 3.6.2c AND 4.6.2c

(a) Each of two differential pressure switches provide inputs to one instrument channel in each trip system.

(b) May be bypassed in the cold shutdown condition.

INSERT 3

(c) ~~Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2c, the primary sensor will be calibrated and tested once per operating cycle.~~

(d) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

(e) With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the action required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

(f) With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within 24 hours.
or
2. Take the ACTION required by Specification 3.6.2a for that Parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour
and
2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within 24 hours.
or
b. Take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2c.

TABLE 4.6.2d

INSTRUMENTATION THAT INITIATES CORE SPRAY

Surveillance Requirement

Parameter	**Sensor Check**	**Instrument Channel Test**	**Instrument Channel Calibration**
START CORE SPRAY PUMPS			
(1) High Drywell Pressure	Note 1 ~~Once/day~~	Note 1 ~~Once per 3 months~~(c)	Note 1 ~~Once per 3 months~~(c)
(2) Low-Low Reactor Water Level	Note 1 ~~Once/day~~	Note 1 ~~Once per 3 months~~(c)	Note 1 ~~Once per 3 months~~(c)
OPEN CORE SPRAY DISCHARGE VALVES			
(3) Reactor Pressure and either (1) or (2) above	None	Note 1 ~~Once per 3 months~~(c)	Note 1 ~~Once per 3 months~~(c)

NOTES FOR TABLES 3.6.2d AND 4.6.2d

(a) May be bypassed when necessary for containment inerting.

(b) May be bypassed when necessary for performing major maintenance as specified in Specification 2.1.1.e.

(c) ~~Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2d, the primary sensor will be calibrated and tested once per operating cycle.~~ INSERT 3

(d) May be bypassed when necessary for integrated leak rate testing.

(e) The instrumentation that initiates the Core Spray System is not required to be operable, if there is no fuel in the reactor vessel.

(f) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2d.

TABLE 4.6.2e

INSTRUMENTATION THAT INITIATES CONTAINMENT SPRAY

Surveillance Requirement

		Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	a.	High Drywell Pressure	~~Once/day~~ Note 1	~~Once per 3 months~~(b) Note 1	~~Once per 3 months~~(b) Note 1
	b.	Low-Low Reactor Water Level	~~Once/day~~ Note 1	~~Once per 3 months~~(b) Note 1	~~Once per 3 months~~(b) Note 1

NOTES FOR TABLES 3.6.2e AND 4.6.2e

(a) May be bypassed in the shutdown mode whenever the reactor coolant temperature is less than 215°F.

INSERT 3

(b) ~~Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2e, the primary sensor will be calibrated and tested once per operating cycle.~~

(c) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip system in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2e.

TABLE 4.6.2f

INSTRUMENTATION THAT INITIATES AUTO DEPRESSURIZATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
INITIATION			
(1) a. Low-Low-Low Reactor Water	None	~~Once per 3 months~~(c) Note 1	~~Once per 3 months~~(c) Note 1
and			
b. High Drywell Pressure	~~Once/day~~ Note 1	~~Once per 3 months~~(c) Note 1	~~Once per 3 months~~(c) Note 1

NOTES FOR TABLES 3.6.2f AND 4.6.2f

(a) Both instrument channels in either trip system are required to be energized to initiate auto depressurization. One trip system is powered from power board 102 and the other trip system from power board 103.

(b) May be bypassed when the reactor pressure is less than 110 psig and the reactor coolant temperature is less than the corresponding saturation temperature.

INSERT 3

(c) ~~Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2f, the primary sensor will be calibrated and tested once per operating cycle.~~

(d) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2f.

TABLE 4.6.2g

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Deleted			
(2)	IRM			
	a. Detector not in Startup Position	N/A	~~Once per week~~[g] Note 1	N/A
	b. Inoperative	N/A	~~Once per week~~[g] Note 1	N/A
	c. Downscale	N/A	~~Once per week~~[g] Note 1	~~Once per operating cycle~~[j] Note 1
	d. Upscale	N/A	~~Once per week~~[g] Note 1	~~Once per operating cycle~~[j] Note 1
(3)	APRM			
	a. Inoperative	None	~~Once per 3 months~~ Note 1	None
	b. Upscale (Biased by Recirculation Flow)	None	~~Once per 3 months~~ Note 1	~~Once per 3 months~~[j] Note 1
	c. Downscale	None	~~Once per 3 months~~ Note 1	~~Once per 3 months~~[j] Note 1
(4)	Deleted			

TABLE 4.6.2g (cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD WITHDRAWAL BLOCK

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(5)	Refuel Platform and Hoists	---	(see 4.5.2)	---
(6)	Mode Switch in Shutdown	---	Note 1 ~~Once during each major refueling outage~~	---
(7)	Mode Switch in Refuel (Blocks withdrawal of more than 1 rod)	---	~~Once during each major refueling outage~~ Note 1	---
(8)	Deleted			

NOTES FOR TABLES 3.6.2g and 4.6.2g

(a) Deleted

(b) No more than one of the four IRM inputs to each instrument channel shall be bypassed. These signals may be bypassed when the APRMs are onscale.

(c) No more than one of the four APRM inputs to each instrument channel shall be bypassed provided that the APRM in the other instrument channel in the same core quadrant is not bypassed. No more than two C or D level LPRM inputs to an APRM shall be bypassed and only four LPRM inputs to only one APRM shall be bypassed in order for the APRM to be considered operable. In the Run mode of operation, bypass of two chambers from one radial core location in any one APRM shall cause that APRM to be considered inoperative. A Travelling In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing. If one APRM in a quadrant is bypassed and meets all requirements for operability with the exception of the requirement of at least one operable chamber at each radial location, it may be returned to service and the other APRM in that quadrant may be removed from service for test and/or calibration only if no control rod is withdrawn during the calibration and/or test.

(d) Deleted

(e) Deleted

(f) One sensor provides input to each of two instrument channels. Each instrument channel is in a separate trip system.

(g) Within 24 hours before startup, if not performed within the previous 7 days. Not required to be performed during shutdown until 12 hours after entering startup from run.

(h) The actuation of either or both trip systems will result in a rod block.

(i) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the Trip System in the tripped condition, provided at least one other operable channel in the same Trip System is monitoring that Parameter.

(j) Neutron detectors are excluded.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2g.

TABLE 4.6.2h

VACUUM PUMP ISOLATION

Surveillance Requirement

Parameter	**Sensor Check**	**Instrument Channel Test**	**Instrument Channel Calibration**
MECHANICAL VACUUM PUMP			
High Radiation Main Steam Line	$\boxed{\text{Note 1}}$ ~~Once/shift~~	$\boxed{\text{Note 1}}$ ~~Once per 3 months~~	$\boxed{\text{Note 1}}$ ~~Once per 3 months~~

(a) Deleted.

(b) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for one trip system, either

1. Place the inoperable channel(s) in the tripped condition within 12 hours.

or

2. Take the ACTION required by Specification 3.6.2a for that Parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement for both trip systems,

1. Place the inoperable channel(s) in one trip system in the tripped condition within one hour.

and

2. a. Place the inoperable channel(s) in the remaining trip system in the tripped condition within 12 hours.

   or

   b. Take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2h.

TABLE 4.6.2i

DIESEL GENERATOR INITIATION

Surveillance Requirements

Parameter	Sensor Check	Instrument[a] Channel Test	Instrument[b] Channel Calibration
Loss of Power			
a. 4.16kV PB 102/103 Emergency Bus Undervoltage (Loss of Voltage)	NA	Note 1 ~~Once per month~~	Note 1 ~~Once per refueling cycle~~
b. 4.16kV PB 102/103 Emergency Bus Undervoltage (Degraded Voltage)	NA	Note 1 ~~Once per month~~	Note 1 ~~Once per refueling cycle~~

(a) The instrument channel test demonstrate the operability of the instrument channel by simulating an undervoltage condition to verify that the tripping logic functions properly.

(b) The instrument channel calibration will demonstrate the operability of the instrument channel by simulating an undervoltage condition to verify that the tripping logic functions properly. In addition, a sensor calibration will be performed to verify the set points listed in Table 3.6.2.i.

(c) A channel may be placed in an inoperable status for up to 2 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2i

TABLE 4.6.2j

EMERGENCY VENTILATION INITIATION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	High Radiation Reactor Building Ventilation Duct	~~Once/shift~~ Note 1	~~Once during each operating cycle~~ Note 1	~~Once per quarter~~ Note 1
(2)	High Radiation Refueling Platform	~~(b)~~ Note 1	(c)	~~Once per quarter~~ Note 1

NOTES FOR TABLES 3.6.2j AND 4.6.2j

(a) This function shall be operable whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building, and ~~during operations~~ during operations with a potential for draining the reactor vessel (OPDRVs).

Deleted

(b) ~~Once per shift whenever this function is required to be operable.~~

INSERT 1

(c) Immediately prior to when function is required and ~~once per week~~ thereafter until function is no longer required.

(d) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one Operable Instrument Channel in the same Trip System is monitoring that parameter.

With the number of Operable channels one less than required by the Minimum Number of Operable Instrument Channels for the Operable Trip System, either

1) Place the inoperable channel(s) in the tripped condition within 24 hours.
or
2) Take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2j

TABLE 4.6.2k

HIGH PRESSURE COOLANT INJECTION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Low Reactor Water Level	~~Once per day~~ Note 1	~~Once per 3 months~~(b) Note 1	~~Once per 3 months~~(b) Note 1
(2)	Automatic Turbine Trip	None	~~Once during each operating cycle~~ Note 1	None

NOTES FOR TABLES 3.6.2k AND 4.6.2k

(a) May be bypassed when the reactor pressure is less than 110 psig and the reactor coolant temperature is less than the corresponding saturation temperature.

INSERT 3

(b) ~~Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2k, the primary sensor will be calibrated and tested once per operating cycle.~~

(c) A channel may be placed in an inoperable status for up to 6 hours for required surveillances without placing the Trip System in the tripped condition provided at least one operable channel in the same Trip System is monitoring that parameter.

With the number of Operable channels less than required by the Minimum Number of Operable Instrument Channels per Operable Trip System requirement:

1. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or take the ACTION required by Specification 3.6.2a for that Parameter.

2. With more than one channel inoperable, take the ACTION required by Specification 3.6.2a for that Parameter.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2k

TABLE 4.6.2I

CONTROL ROOM AIR TREATMENT SYSTEM INITIATION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Low-Low Reactor Water Level	Note 1 ~~Once/shift~~	Note 1 ~~Once per quarter~~[(b)]	Note 1 ~~Once per quarter~~[(b)]
(2)	High Steam Flow Main-Steam Line	Note 1 ~~Once/day~~	Note 1 ~~Once per quarter~~[(b)]	Note 1 ~~Once per quarter~~[(b)]
(3)	High Temperature Main-Steam Line Tunnel	---	Note 1 ~~Once each operating cycle not to exceed 24 months~~	Note 1 ~~Once each operating cycle not to exceed 24 months~~
(4)	High Drywell Pressure	Note 1 ~~Once/day~~	Note 1 ~~Once per quarter~~[(b)]	Note 1 ~~Once per quarter~~[(b)]

NOTES FOR TABLES 3.6.2I AND 4.6.2I

(a) May be bypassed when necessary for containment inerting.

INSERT 3

(b) ~~Only the trip circuit will be calibrated and tested at the frequencies specified; the primary sensor will be calibrated and tested once per operating cycle.~~

(c) May be bypassed in the cold shutdown condition.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.2I

High Flow-Main Steam Line, ±1 psid

High Flow-Emergency Cooling Line, ±1 psid

High Area Temperature-Main Steam Line, ±10°F

High Area Temperature-Clean-up and Shutdown, ±6°F

High Radiation-Main Steam Line, +100% and -50% of set point value

High Radiation-Reactor Building Vent, +100% and -50% of set point

High Radiation-Refueling Platform, +100% and -50% of set point

~~Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," MDE-77-0485, "Technical Specification Improvement Analysis for Nine Mile Point Nuclear Station, Unit 1," and Generic Letter 91-04, "Changes in Technical Specification Intervals to Accommodate a 24-Month Fuel Cycle."~~

~~Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A Suppl 2, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," and with NEDC-31677P-A, "Technical Specification Improvement Analyses for BWR Isolation Actuation Instrumentation." Because of local high radiation, testing instrumentation in the area of the main steam line isolation valves can only be done during periods of Station shutdown. These functions include high area temperature isolation and isolation valve position scram.~~

~~Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation)," Parts 1 and 2 and RE-003, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Nine Mile Point Nuclear Station, Unit 1."~~

~~Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992).~~

INSERT 3

Testing of the scram associated with the shutdown position of the mode switch can be done only during periods of Station shutdown since it always involves a scram.

b. The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR is maintained greater than the SLCPR. The trip logics for these functions are 1 out of n; e.g., any trip on one of the eight APRM's, eight IRM's or four SRM's will result in a rod block. The minimum instrument channel requirements provide sufficient instrumentation to assure the single failure criteria is met. ~~Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A Suppl 1, "Technical Specification Improvement Analyses for BWR Control Rod Block Instrumentation," GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992), and Generic Letter 91-04, "Changes in Technical Specification Intervals to Accommodate a 24 Month Fuel Cycle."~~

INSERT 3

The APRM rod block trip is flow biased and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the SLCPR.

The APRM rod block also provides local protection of the core; i.e., the prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked before the MCPR reaches the SLCPR, thus allowing adequate margin. Below ~60% power the worst case withdrawal of a single control rod results in a MCPR > SLCPR without rod block action, thus below this level it is not required.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the SLCPR.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.3 EMERGENCY POWER SOURCES

Applicability:

Applies to the operational status of the emergency power sources.

Objective:

To assure the capability of the emergency power sources to provide the power required for emergency equipment in the event of a loss-of-coolant accident.

Specification:

a. For all reactor operating conditions except cold shutdown, there shall normally be available two 115 kv external lines, two diesel generator power systems and two battery systems, except as further specified in "b," "c," "d," "e," and "h" below.

b. One 115 kv external line may be de-energized provided two diesel-generator power systems are operable. If a 115 kv external line is de-energized, that line shall be returned to service within 7 days.

4.6.3 EMERGENCY POWER SOURCES

Applicability:

Applies to the periodic testing requirements for the emergency power sources.

Objective:

To assure the operability of the emergency power sources to provide emergency power required in the event of a loss-of-coolant accident.

Specification:

The emergency power systems surveillance will be performed as indicated below. In addition, components on which maintenance has been performed will be tested.

INSERT 1

a. ~~During each major refueling outage~~ - test for automatic startup and pickup of load required for a loss-of-coolant accident.

INSERT 1

b. ~~Monthly~~ - manual start and operation at rated load shall be performed for a minimum time of one hour. Determine the specific gravity of each cell. Determine the battery voltage.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. One diesel-generator power system may be inoperable provided two 115 kv external lines are energized. If a diesel-generator power system becomes inoperable, it shall be returned to an operable condition within 14 days. In addition, if a diesel-generator power system becomes inoperable coincident with a 115 kv line de-energized, that diesel-generator power system shall be returned to an operable condition within 24 hours.	INSERT 1 c. ~~Weekly –~~ determine the cell voltage and specific gravity of the pilot cells of each battery.
d. If a reserve power transformer becomes inoperable, it shall be returned to service within seven days.	d. Surveillance for startup with an inoperable diesel-generator – prior to startup the operable diesel-generator shall be tested for automatic startup and pickup of the load required for a loss-of-coolant accident.
e. For all reactor operating conditions except startup and cold shutdown, the following limiting conditions shall be in effect: (1) One operable diesel-generator power system and one energized 115 kv external line shall be available. If this condition is not met, normal orderly shutdown will be initiated within one hour and the reactor will be in the cold shutdown condition within ten hours.	e. Surveillance for operation with an inoperable diesel-generator – If a diesel-generator becomes inoperable from any cause other than an inoperable support system or preplanned maintenance or testing, within 8 hours, either determine that the cause of the diesel-generator being inoperable does not impact the operability of the operable diesel-generator or demonstrate operability by testing the operable diesel-generator. Operability by testing will be demonstrated by achieving steady state voltage and frequency.

If a battery system becomes inoperable, that battery system must be returned to an operable status within 24 hours. If the 24-hour allowed outage time cannot be met, then Specification 3.0.1 must be entered immediately. The second paragraph of Specification 3.0.1 provides two options:

1. Place the unit in a condition consistent with the individual specification; however, in this case, the individual specification (i.e., 3.6.3) does not provide any action to take when a battery system has been inoperable for more than 24 hours. To determine required actions and action completion times, the individual specifications for the systems supported by the battery system should be entered and reviewed to determine applicable actions. If no actions are applicable for the given reactor operating condition, then no actions are required.

or

2. Place the unit in an operational condition in which the specification is not applicable (i.e., cold shutdown).

INSERT 3

LIMITING CONDITION FOR OPERATION

3.6.5 Radioactive Material Sources

Applicability:

Applies to the limit on source leakage for sealed or start-up sources.

Objective:

To specify the requirements necessary to limit contamination from radioactive source materials.

Specification:

1. The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. If the test reveals the presence of 0.005 microcurie or more of removable contamination, it shall immediately be withdrawn from use, decontaminated and repaired or be disposed of in accordance with Commission regulations. Sealed sources are exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material or 10 microcuries or less of alpha emitting material.

2. Results of required leak tests performed on sources, if the tests reveal the presence of 0.005 microcurie or more of removable contamination, shall be reported within 90 days.

SURVEILLANCE REQUIREMENTS

4.6.5 Radioactive Material Sources

Applicability:

Applies to the periodic testing requirements for source leakage.

Objective:

To assure the capability of each source material container to limit leakage within allowable limits.

Specification:

Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

1. Each sealed source, except start-up sources subject to core flux, containing radioactive material, other than hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination ~~at intervals not to exceed six months.~~

INSERT 1

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3. A complete inventory of radioactive by-product materials, exceeding the limits set forth in 10 CFR 30.71, in sealed sources in possession shall be maintained current at all times.	2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested. INSERT 1 3. Start-up sources shall be leak tested ~~within 31 days~~ prior to being subjected to core flux and following any repair or maintenance.

BASES FOR 3.6.5 AND 4.6.5 RADIOACTIVE MATERIAL SOURCES

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the probable leakage from the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Quantities of interest to this specification which are exempt from the leakage testing are consistent with the criteria of 10 CFR Parts 30.11-20 and 70.19. Leakage from sources excluded from the requirements of this specification is not likely to represent more than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.11 ACCIDENT MONITORING INSTRUMENTATION

Applicability:

Applies to the operability of the plant instrumentation that performs an accident monitoring function.

Objective:

To assure high reliability of the accident monitoring instrumentation.

Specification:

a. During the power operating condition, the accident monitoring instrumentation channels shown in Table 3.6.11-1 shall be operable except as specified in Table 3.6.11-2.

4.6.11 ACCIDENT MONITORING INSTRUMENTATION

Applicability:

Applies to the surveillance of the instrumentation that performs an accident monitoring function.

Objective:

To verify the operability of accident monitoring instrumentation.

Specification:

Instrument channels shall be tested and calibrated ~~at least as frequently as listed in Table 4.6.11.~~

at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.11

TABLE 4.6.11

ACCIDENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

Parameter	Instrument Channel Test	Instrument Channel Calibration
(1) Deleted		
(2) Deleted		
(3) Reactor vessel water level	Note 1 ~~Once per quarter~~	Note 1 ~~Once during each major refueling outage~~
(4) Drywell Pressure Monitor	Note 1 ~~Once per month~~	Note 1 ~~Once during each major refueling outage~~
(5) Suppression Chamber Water Level Monitor	Note 1 ~~Once per quarter~~	Note 1 ~~Once during each major refueling outage~~
(6) Deleted		
(7) Containment High Range Radiation Monitor	Note 1 ~~Once per month~~	Note 1 ~~Once during each major refueling outage~~
(8) Suppression Chamber Water Temperature	Note 1 ~~Once per month~~	Note 1 ~~Once during each major refueling outage~~

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program.

Accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980, NUREG-0661, "Safety Evaluation Report Mark I Containment Long Term Program," and the NRC Final Rule, "Combustible Gas Control in Containment," made effective October 16, 2003 (68 FR 54123).

~~Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out Of Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992).~~

Action 3a and Action 4a of Table 3.6.11-2 require that with the number of OPERABLE channels less than the total Number of Channels shown in Table 3.6.11-1, a Special Report must be prepared and submitted to the NRC within 14 days following the event. The term "event" refers to the reason that an instrument channel is inoperable. For the purpose of applying Action 3a and Action 4a of Table 3.6.11-2, removal of a single accident monitoring instrumentation channel from service for the sole purpose of performing routine TS required surveillances is not considered an event requiring preparation and submittal of a 14-day Special Report. If a single accident monitoring instrumentation channel is removed from service for other activities (e.g., to perform preventive maintenance), or if a channel fails, these events require preparation and submittal of a Special Report in accordance with Actions 3a and 4a.

INSERT 3

LIMITING CONDITION FOR OPERATION

3.6.12 REACTOR PROTECTION SYSTEM AND REACTOR TRIP SYSTEM POWER SUPPLY MONITORING

Applicability:

Applies to the operability of instrumentation that provides protection of the reactor protection system and reactor trip system.

Objective:

To assure the operability of the instrumentation monitoring the power to the reactor protection system and reactor trip system.

Specification:

a. Except as specified in specifications b and c below, two protective relay systems shall be operable for each power supply.

SURVEILLANCE REQUIREMENT

4.6.12 REACTOR PROTECTION SYSTEM AND REACTOR TRIP SYSTEM POWER SUPPLY MONITORING

Applicability:

Applies to the surveillance of instrumentation that provides protection of the reactor protection system and reactor trip system.

Objective:

To verify the operability of protection instrumentation monitoring the power to the reactor protection and reactor trip buses.

Specification: INSERT 1

a. ~~At least once every six months~~ Demonstrate operability of the overvoltage, undervoltage and underfrequency protective instrumentation by performing an instrument channel test. This instrument channel test will consist of simulating abnormal power conditions by applying from a test source, an overvoltage signal, an undervoltage signal and an under-frequency signal to verify that the tripping logic up to but not including the output contactors functions properly.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. With one protective relaying system inoperable, restore the inoperable system to an operable status within 72 hours or remove the power supply from service. c. With both protective relaying systems inoperable, restore at least one to an operable status within 30 minutes or remove the power supply from service.	INSERT 1 b. ~~At least once per refueling cycle~~ Demonstrate operability of the overvoltage, undervoltage and underfrequency protective instrumentation by performing an instrument channel test. This instrument channel test will consist of simulating abnormal power conditions by applying from a test source an overvoltage signal, an undervoltage signal and an underfrequency signal to verify that the tripping logic including the output contactors functions properly at least once. In addition, a sensor calibration will be performed to verify the following setpoints. i. Overvoltage ≤132 volts, ≤4 seconds ii. Undervoltage ≥108 volts, ≤4 seconds iii. Underfrequency ≥ 57 hertz, ≤2 seconds

BASES FOR 3.6.12 AND 4.6.12 REACTOR PROTECTION SYSTEM AND REACTOR TRIP SYSTEM POWER SUPPLY MONITORING

To eliminate the potential for undetectable single component failure which could adversely affect the operability of the reactor protection system and reactor trip system, protective relaying schemes are installed on Motor Generator Sets 131 and 141, Static Uninterruptible Power Supply Systems 162 and 172, and maintenance bus 130A. This provides for overvoltage, undervoltage and underfrequency protection.

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.13 REMOTE SHUTDOWN PANELS

Applicability:

Applies to the operating status of the remote shutdown panels.

Objective:

To assure the capability of the remote shutdown panels to provide 1) initiation of the emergency condensers independent of the main/auxiliary control room 2) control of the motor-operated steam supply valves independent of the main/auxiliary control room and 3) parameter monitoring outside the control room.

Specification:

a. During power operation, the remote shutdown panels' Functions in Table 3.6.13-1 shall be operable.

4.6.13 REMOTE SHUTDOWN PANELS

Applicability:

Applies to the periodic testing requirements for the remote shutdown panels.

Objective:

To assure the capability of the remote shutdown panels to provide 1) initiation of the emergency condensers independent of the main/auxiliary control room 2) control of the motor-operated steam supply valves independent of the main/auxiliary control room and 3) parameter monitoring outside the control room.

Specification:

The remote shutdown panels surveillance shall be performed as indicated below:

a. Each remote shutdown panel monitoring instrumentation channel shall be demonstrated operable by performance of the operations and ~~frequencies shown in Table 4.6.13-1.~~

at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.13-1

INSERT 1

b. ~~During each major refueling outage~~

1. Each remote shutdown panel shall be demonstrated to initiate the emergency condensers independent of the main/auxiliary control room.

TABLE 4.6.13-1

REMOTE SHUTDOWN PANEL MONITORING

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Calibration
Reactor Pressure	Note 1 ~~Once per day~~	Note 1 ~~Once per 3 months~~(a)
Reactor Water Level	Note 1 ~~Once per day~~	Note 1 ~~Once per 3 months~~(a)
Reactor Water Temperature	Note 1 ~~Once per day~~	Note 1 ~~Once per refueling cycle~~
Torus Water Temperature	Note 1 ~~Once per day~~	Note 1 ~~Once per refueling cycle~~
Drywell Pressure	Note 1 ~~Once per day~~	Note 1 ~~Once per 3 months~~(a)
Emergency Condenser Water Level	Note 1 ~~Once per day~~	Note 1 ~~Once per refueling cycle~~
Drywell Temperature	Note 1 ~~Once per day~~	Note 1 ~~Once per refueling cycle~~
"All Rods In" Light	Note 1 ~~Once per refueling cycle~~	N/A

(a) The indicator located at the remote shutdown panel will be calibrated at the frequency listed in Table 4.6.13-1. Calibration of the remaining channel instrumentation is provided by Specification 4.6.2.

Note 1: Surveillance intervals are specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.6.13-1

The remote shutdown panels provide 1) manual initiation of the emergency condensers 2) manual control of the steam supply valves and 3) parameters monitoring independent of the main/auxiliary control room. Two panels are provided, each located in a separate fire area, for added redundancy. Both panels are also in separate fire areas from the main/auxiliary control room. One channel of each Function provides the necessary capabilities consistent with 10 CFR 50.48(c). Therefore, only one channel of either remote shutdown panel monitoring instrument or control is required to be operable. The electrical design of the panels is such that no single fire can cause loss of both emergency condensers.

Each remote shutdown panel is provided with controls for one emergency condenser loop. The emergency condensers are designed such that automatic initiation is independently assured in the event of a fire 1) in the Reactor Building (principle relay logic located in the auxiliary control room or 2) in the main/auxiliary control room or Turbine Building (redundant relay logic located in the Reactor Building). Each remote shutdown panel also has controls to operate the two motor-operated steam supply valves on its respective emergency condenser loop. A key operated bypass switch is provided to override the automatic isolation signal to these valves. Once the bypass switch is activated, the steam supply valves can be manually controlled from the remote shutdown panels. Since automatic initiation of the emergency condenser is assured, the remote shutdown panels serve as additional manual controlling stations for the emergency condensers. In addition, certain parameters are monitored at each remote shutdown panel.

The remote shutdown panels are normally de-energized, except for the monitoring instrumentation, which is normally energized. To energize the remaining functions on a remote shutdown panel, a power switch located on each panel must be activated. Once the panels are completely energized, the emergency condenser condensate return valve and steam supply valve controls can be utilized.

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.15 MAIN CONDENSER OFFGAS	4.6.15 MAIN CONDENSER OFFGAS
Applicability: Applies to the radioactive effluents from the main condenser.	Applicability: Applies to the periodic test and recording requirements of main condenser offgas.
Objective: To assure that radioactive material is not released to the environment in any uncontrolled manner and is within the limits of 10CFR20 and 10CFR50, Appendix I.	Objective: To ascertain that radioactive effluents from the main condenser are within allowable values of 10CFR20, Appendix B and 10CFR50, Appendix I.
Specification: The gross radioactivity (beta and/or gamma) rate of noble gases measured at the recombiner discharge shall be limited to less than or equal to 500,000 μCi/sec. This limit can be raised to 1 Ci/sec. for a period not to exceed 60 days provided the offgas treatment system is in operation. With the gross radioactivity (beta and/or gamma) rate of noble gases at the recombiner discharge exceeding the above limits, restore the gross radioactivity rate to within its limit within 72 hours or be in at least Hot Shutdown within the next 12 hours.	Specification: The gross radioactivity (beta and/or gamma) rate of noble gases from the recombiner discharge shall be determined to be within the limits of Specification 3.6.15 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the recombiner discharge: INSERT 1 ~~Monthly.~~ Within 4 hours following an increase on the recombiner discharge monitor of greater than 50%, factoring out increases due to changes in thermal power level and dilution flow changes.

Restricting the gross radioactivity rate of noble gases from the main condenser provides assurance that the total effective dose equivalent to an individual at the exclusion area boundary will not exceed a very small fraction of the limits of 10 CFR 50.67 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50. The primary purpose of providing this specification is to limit buildup of fission product activity within the station systems which would result if high fuel leakage were to be permitted over extended periods.

INSERT 3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.7.1 SPECIAL TEST EXCEPTION - SHUTDOWN MARGIN DEMONSTRATIONS

Applicability:

Applies to shutdown margin demonstration in the cold shutdown condition.

Objective:

To assure the capability of the control rod system to control core reactivity.

a. The reactor mode switch may be placed in the startup position to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

   (1) The source range monitors are operable in the noncoincident condition.

   (2) The rod worth minimizer is operable per Specification 3.1.1b(3)(b) and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

4.7.1 SPECIAL TEST EXCEPTION - SHUTDOWN MARGIN DEMONSTRATIONS

Applicability:

Applies to periodic inspections required to perform shutdown margin demonstrations in the cold shutdown condition.

Objective:

To specify the inspections required to perform the shutdown margin demonstration in the cold shutdown condition.

INSERT 1

a. Within 30 minutes prior to and ~~at least once per 12 hours~~ during the performance of a shutdown margin demonstration, verify that:

   (1) The source range monitors are operable per Specification 3.5.1.

   (2) The rod worth minimizer is operable with the required program per Specification 3.1.1b(3)(b) or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown margin demonstration procedure.

The shutdown margin demonstration may be performed prior to power operation. However, the mode switch must be placed in the startup position to allow withdrawal of more than one control rod. Specifications 3.7.1 and 4.7.1 require certain restrictions in order to ensure that an inadvertent criticality does not occur while performing the shutdown margin demonstration.

This special test exception provides the appropriate additional controls to allow the shutdown margin demonstration to be performed in the cold shutdown condition with the vessel head in place. Compliance with this special test exception is optional and applies only if the shutdown margin demonstration will be performed prior to the reactor coolant system pressure and control rod scram time tests following refueling outages when core alterations are performed. The shutdown margin demonstration is performed using the in-sequence non-critical method.

INSERT 3

e. The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the 10 CFR 50 Appendix J Testing Program Plan.

The provisions of Specification 4.0.3 are applicable to the 10 CFR 50 Appendix J Testing Program Plan.

6.5.8 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Air Treatment (CRAT) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation of the CRAT System, operating at a flow rate of 2025-2475 cfm, at a Frequency of 24 months. 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.

f. The provisions of TS 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.

INSERT 2

ATTACHMENT 4

License Amendment Request

Nine Mile Point Nuclear Station, Unit 1
Docket No. 50-220

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

TSTF-425 (NUREG-1433) vs. NMP Unit 1 Cross-Reference

TSTF-425 (NUREG-1433) vs. NMP Unit 1 Cross-Reference

Technical Specification Section Title/Surveillance Description*	TSTF-425	NMP-1
Control Rod Operability	**3.1.3**	**3.1.1**
Control rod position	3.1.3.1	------------
Notch test - fully withdrawn control rod one notch	3.1.3.2	4.1.1.(2)
Notch test - partially withdrawn control rod one notch	3.1.3.3	4.1.1.(2)
Control Rod Scram Times	**3.1.4**	------------
Scram time testing	3.1.4.2	4.1.1.c(3)
Control Rod Scram Accumulators	**3.1.5**	------------
Control rod scram accumulator pressure	3.1.5.1	4.1.1.d
Rod Pattern Control	**3.1.6**	-----------
[BPWS] Analyzed rod position sequence	3.1.6.1	-----------
Standby Liquid Control (SLC) System	**3.1.7**	**3.1.2**
Volume of sodium pentaborate	3.1.7.1	4.1.2.b.2
Temperature of sodium pentaborate solution	3.1.7.2	4.1.2.b.3
Temperature of pump suction piping	3.1.7.3	-----------
Continuity of explosive charge	3.1.7.4	4.1.2.a.1
Concentration of boron solution	3.1.7.5	4.1.2.b.1
Manual/power operated valve position	3.1.7.6	-------------
Pump flow rate	3.1.7.7	4.1.2.a.2
Flow through one SLC subsystem	3.1.7.8	4.1.2.a.1
Heat traced piping is unblocked	3.1.7.9	------------
Verify sodium pentaborate enrichment [Solution Boron-10 Enrichment] (NUREG-1433 - prior to addition to SLC tank)	3.1.7.10	4.1.2.b.4
Surveillances with inoperable components	-----------	4.1.2.c
Scram Discharge Volume (SDV) Vent & Drain Valves	**3.1.8**	-----------
Each SDV vent & drain valve open	3.1.8.1	4.1.1.e(1)
Cycle each SDV vent & drain valve fully closed/fully open position	3.1.8.2	4.1.1.e(2)
Each SDV vent & drain valve closes on receipt of scram	3.1.8.3	4.1.1.e(1)
Average Planar Linear Heat Generation Rate (APLHGR)	**3.2.1**	-----------
APLHGR less than or equal to limits	3.2.1.1	4.1.7.a
Minimum Critical Power Ratio (MCPR)	**3.2.2**	-----------
MCPR greater than or equal to limits	3.2.2.1	4.1.7.c
Linear Heat Generation Rate (LHGR)	**3.2.3**	----------
LHGR less than or equal to limits	3.2.3.1	4.1.7.b
Power Flow Relationship	-----------	4.1.7.d
Average Power Range Monitor (APRM) Gain & Setpoints	**3.2.4**	-----------
MFLPD is within limits	3.2.4.1	4.6.2.c
APRM setpoints or gain are adjusted for calculated MFLPD	3.2.4.2	4.6.2.c
Reactor Protection System (RPS) Instrumentation	**3.3.1.1**	**3.6.2**
Channel Check	3.3.1.1.1	Per Table 4.6.2.a
Absolute diff. between APRM channels & calculated power	3.3.1.1.2	4.6.2.c
Adjust channel to conform to calibrated flow (APRM STP – Hi)	3.3.1.1.3	Table 4.6.2.a
Channel Functional Test (12 hours after entering Mode 2)	3.3.1.1.4	-----------
Channel Functional Test (weekly/monthly)	3.3.1.1.5	-----------

Technical Specification Section Title/Surveillance Description*	**TSTF-425**	**NMP-1**
Calibrate local power range monitors	3.3.1.1.6	Table 4.6.2.a
Channel Functional Test	3.3.1.1.7	----------
Calibrate trip units (quarterly)	3.3.1.1.8	----------
Channel Calibration (APRMs)	3.3.1.1.9	Table 4.6.2.a
Channel Functional Test (Reactor Mode Switch)	3.3.1.1.10	----------
Channel Calibration	3.3.1.1.11	Per Table 4.6.2.a
Verify APRM Flow Biased STP – High	3.3.1.1.12	Table 4.6.2.a
Logic System Functional Test	3.3.1.1.13	----------
Verify TSV/TCV closure/Trip Oil Press-Low Not Bypassed	3.3.1.1.14	Table 4.6.2.a
Verify RPS Response Time	3.3.1.1.15	----------
Source Range Monitor (SRM)	**3.3.1.2**	----------
Channel Check	3.3.1.2.1	----------
Verify Operable SRM Detector	3.3.1.2.2	----------
Channel Check	3.3.1.2.3	----------
Verify count rate	3.3.1.2.4	----------
Channel Functional Test (7 days)	3.3.1.2.5	----------
Channel Functional Test (31 days)	3.3.1.2.6	----------
Channel Calibration	3.3.1.2.7	----------
Control Rod Block Instrumentation	**3.3.2.1**	**3.6.2**
Channel Functional Test (routine)	3.3.2.1.1	----------
Channel Functional Test (rod withdrawal at $\leq$ 10% RTP)	3.3.2.1.2	----------
Channel Functional Test (thermal power $\leq$ 10%)	3.3.2.1.3	----------
Verify RBM	3.3.2.1.4	----------
Verify RWM not bypassed	3.3.2.1.5	----------
Channel Functional Test	3.3.2.1.6	Table 4.6.2g
Channel Calibration	3.3.2.1.7	----------
Feedwater & Main Turbine High Water Level Trip Instrumentation	**3.3.2.2**	----------
Channel Check	3.3.2.2.1	----------
Channel Functional Test	3.3.2.2.2	----------
Channel Calibration	3.3.2.2.3	----------
Logic System Functional Test	3.3.2.2.4	----------
Post Accident Monitor (PAM) Instrumentation	**3.3.3.1**	**3.6.11**
Channel Check	3.3.3.1.1	Per Table 4.6.11
Calibration	3.3.3.1.2	Per Table 4.6.11
Remote Shutdown System [Alternate Shutdown Monitoring]	**3.3.3.2**	**3.6.13**
Channel Check	3.3.3.2.1	Per Table 4.6.13 -1
Verify control circuit and transfer switch capable of function	3.3.3.2.2	----------
Channel Calibration	3.3.3.2.3	Per Table 4.6.13 -1
End-of-Cycle-Recirculation Pump Trip (RPT) Instrumentation	**3.3.4.1**	----------
Channel Functional Test	3.3.4.1.1	----------
Calibrate trip units	3.3.4.1.2	----------
Channel Calibration	3.3.4.1.3	----------

Technical Specification Section Title/Surveillance Description*	**TSTF-425**	**NMP-1**
Logic System Functional Test	3.3.4.1.4	----------
Verify TSV/TCV Closure/Trip Oil Press-Low Not Bypassed	3.3.4.1.5	----------
Verify EOC-RPT System Response Time	3.3.4.1.6	----------
Determine RPT breaker interruption time	3.3.4.1.7	----------
Anticipated Trip Without Scram-RPT Instrumentation	**3.3.4.2**	----------
Channel Check	3.3.4.2.1	----------
Channel Functional Test	3.3.4.2.2	----------
Calibrate trip units	3.3.4.2.3	----------
Channel Calibration	3.3.4.2.4	----------
Logic System Functional Test	3.3.4.2.5	----------
Emergency Core Cooling System (ECCS) Instrumentation	**3.3.5.1**	----------
Channel Check	3.3.5.1.1	**Tables 4.6.2.d and f**
Channel Functional Test	3.3.5.1.2	----------
Calibrate trip units	3.3.5.1.3	----------
Channel Calibration (HPCI: Condensate Storage Tank Level - Low)	3.3.5.1.4	----------
Channel Calibration	3.3.5.1.5	**Tables 4.6.2.d and f**
Logic System Functional Test	3.3.5.1.6	----------
Verify ECCS Response Time	3.3.5.1.7	----------
Reactor Core Isolation Cooling (RCIC) System Instrumentation [Isolation Condenser Instrumentation]	**3.3.5.2**	**Table 4.6.2.c**
Channel Check	3.3.5.2.1	**Table 4.6.2.c**
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Calibrate trip units	3.3.5.2.3	----------
Channel Calibration (Condensate Storage Tank Level – Low)	3.3.5.2.4	----------
Channel Calibration	3.3.5.2.5	**Table 4.6.2.c**
Logic System Functional Test	3.3.5.2.6	----------
Primary Containment Isolation Instrumentation	**3.3.6.1**	**Table 4.6.2.b**
Channel Check	3.3.6.1.1	**Table 4.6.2.b**
Channel Functional Test	3.3.6.1.2	----------
Calibrate trip units	3.3.6.1.3	----------
Channel Calibration	3.3.6.1.4	**Table 4.6.2.b**
Channel Functional Test (HPCI/RCIC Suppr. Pool Area Temp.)	3.3.6.1.5	------------
Channel Calibration	3.3.6.1.6	------------
Logic System Functional Test	3.3.6.1.7	------------
Verify Isolation Response Time	3.3.6.1.8	------------
Secondary Containment Isolation Instrumentation	**3.3.6.2**	------------
Channel Check	3.3.6.2.1	------------
Channel Functional Test	3.3.6.2.2	------------
Calibrate trip units	3.3.6.2.3	------------
Channel Calibration (Refueling Floor Exhaust Rad. – High)	3.3.6.2.4	------------
Channel Calibration	3.3.6.2.5	------------
Logic System Functional Test	3.3.6.2.6	------------
Verify Isolation Response Time	3.3.6.2.7	------------

Technical Specification Section Title/Surveillance Description*	TSTF-425	NMP-1
Low-Low-Set (LLS) Instrumentation	**3.3.6.3**	------------
Channel Check	3.3.6.3.1	------------
Channel Functional Test	3.3.6.3.2	------------
Channel Functional Test	3.3.6.3.3	------------
Channel Functional Test	3.3.6.3.4	------------
Calibrate trip units	3.3.6.3.5	------------
Channel Calibration	3.3.6.3.6	------------
Logic System Functional Test	3.3.6.3.7	------------
Main Control Room Environmental Control (MCREC) / Emergency Ventilation Initiation	**3.3.7.1**	**Table 4.6.2 j**
Channel Check	3.3.7.1.1	------------
Channel Functional Test	3.3.7.1.2	------------
Calibrate trip units	3.3.7.1.3	------------
Channel Calibration	3.3.7.1.4	------------
Logic System Functional Test	3.3.7.1.5	------------
Loss of Power (LOP) Instrumentation / Diesel Generator Initiation	**3.3.8.1**	**Table 4.6.2 i**
Channel Check	3.3.8.1.1	------------
Channel Functional Test	3.3.8.1.2	------------
Channel Calibration	3.3.8.1.3	------------
Logic System Functional Test	3.3.8.1.4	------------
RPS Electric Power Monitoring	**3.3.8.2**	**3.6.12**
Channel Functional Test	3.3.8.2.1	4.6.12.a
Channel Calibration (RPS MG set/alt. power supply monitoring)	3.3.8.2.2	4.6.12.b
System functional test	3.3.8.2.3	4.6.12.b
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Recirc loop jet pump flow mismatch with both loops operating	3.4.1.1	------------
Jet Pumps	**3.4.2**	------------
Criteria satisfied for each operating recirc loop	3.4.2.1	------------
Safety/Relief Valves (SRVs)	**3.4.3**	**3.2.9**
Safety function lift setpoints	3.4.3.1	4.2.9.c
SRV opens when manually actuated	3.4.3.2	4.2.9.b
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RCS Pressure Isolation Valve (PIV) Leakage	**3.4.5**	------------
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RCS Leakage Detection Instrumentation	**3.4.6**	**3.2.5**
Channel Check	3.4.6.1	------------
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RCS Specific Activity	**3.4.7**	**3.2.4**
Dose Equivalent I-131 specific activity	3.4.7.1	4.2.4.a
RCS Chemistry	------------	**3.2.3**
Chloride ion content and conductivity	------------	4.2.3
Residual Heat Removal (RHR) Shutdown Cooling - Hot Shutdown	**3.4.8**	------------
One RHR Shutdown cooling subsystem operating	3.4.8.1	------------
RHR Shutdown Cooling - Cold Shutdown	**3.4.9**	------------

Technical Specification Section Title/Surveillance Description*	**TSTF-425**	**NMP-1**
One RHR Shutdown cooling subsystem operating	3.4.9.1	------------
RCS Pressure/Temperature Limit	**3.4.10**	------------
RCS pressure, temperature, heatup and cooldown rates	3.4.10.1	------------
RPV flange/head flange temperatures (tensioning head bolt stud)	3.4.10.7	------------
RPV flange/head flange temperatures (after RCS temp $\leq$ 80°F)	3.4.10.8	------------
RPV flange/head flange temperatures (after RCS temp $\leq$ 100°F)	3.4.10.9	------------
Reactor Steam Dome Pressure	**3.4.11**	------------
Verify reactor steam dome pressure	3.4.11.1	------------
ECCS – Operating / Core Spray and ADS	**3.5.1**	**3.1.4 & 3.1.5**
Verify injection/spray piping filled with water	3.5.1.1	------------
Verify each valve in flow path is in correct position	3.5.1.2	------------
Verify ADS nitrogen pressure	3.5.1.3	------------
Verify RHR (LPCI) cross tie valve is closed and power removed	3.5.1.4	------------
Verify LPCI inverter output voltage	3.5.1.5	------------
Verify ECCS pumps develop specified flow	3.5.1.7	4.1.4.b 4.1.6.a 4.1.6.b
Verify HPCI flow rate (Rx press < [1020], > [920])	3.5.1.8	------------
Verify HPCI flow rate (Rx press < [165])	3.5.1.9	------------
Verify ECCS actuates on initiation signal	3.5.1.10	------------
Verify ADS actuates on initiation signal	3.5.1.11	4.1.5.b
Verify each ADS valve opens [actuator strokes] when manually actuated	3.5.1.12	4.1.5.a
Verify motor-operated valve operability [Core Spray System, Containment Cooling System, Fire Protection System]	------------	4.1.4.c
Verify pump compartment water-tight doors closed [Core Spray System, Containment Cooling System]	------------	4.4.3.a
Verify injection/spray piping filled with water	------------	4.1.4.g
Surveillance with inoperable System	------------	4.1.4.e 4.1.6.c
ECCS - Shutdown	**3.5.2**	------------
Verify, for LPCI, suppression pool water level	3.5.2.1	4.3.2.a
Verify, for CS, suppression pool water level and CST water level	3.5.2.2a 3.5.2.2b	4.1.4.f
Verify ECCS piping filled with water	3.5.2.3	4.1.4.g
Verify each valve in flow path is in correct position	3.5.2.4	------------
Verify each ECCS pump develops flow	3.5.2.5	4.1.4.b
Verify ECCS actuates on initiation signal	3.5.2.6	------------
RCIC System [Isolation Condenser]	**3.5.3**	------------
Verify RCIC piping filled with water	3.5.3.1	------------
Verify each valve in flow path is in correct position	3.5.3.2	------------
Verify RCIC flow rate (Rx press < [1020], > [920])	3.5.3.3	------------
Verify RCIC flow rate (Rx press < [165])	3.5.3.4	------------
Verify RCIC actuates on initiation signal	3.5.3.5	------------
Emergency Cooling System	------------	**3.1.3**
System Heat Removal Capability	------------	4.1.3.a

Technical Specification Section Title/Surveillance Description*	**TSTF-425**	**NMP-1**
Shell Side and Makeup Tank water level	------------	4.1.3.b
Manual open/close makeup tank water level control valve	-----------	4.1.3.c
Area Temperature	------------	4.1.3.d
Surveillance with inoperable System	-----------	4.1.3.f
Primary Containment	**3.6.1.1**	**3.3.0**
Verify drywell to suppression chamber differential pressure	3.6.1.1.2	-----------
Primary Containment Air Lock	**3.6.1.2**	------------
Verify only one door can be opened at a time	3.6.1.2.2	-----------
Primary Containment Isolation Valves (PCIVs)	**3.6.1.3**	**3.3.4**
Verify purge valve is closed except one valve in a penetration	3.6.1.3.1	-----------
Verify each 18 inch (6 inch & 18 inch) PC purge valve is closed	3.6.1.3.2	------------
Verify each manual PCIV outside containment is closed	3.6.1.3.3	-----------
Verify continuity of traversing incore probe (TIP) shear valve	3.6.1.3.5	------------
Verify isolation time of each power operated PCIV	3.6.1.3.6	4.3.4.a
Perform leakage rate testing on each PC purge valve	3.6.1.3.7	-----------
Verify isolation time of MSIVs	3.6.1.3.8	------------
Verify automatic PCIV actuates to isolation position	3.6.1.3.9	4.3.4.a
Cycle normally open isolation MOVs	-----------	4.3.4.b
Verify sample of Excess Flow Check Valves actuate to isolation position	3.6.1.3.10	4.3.4.c
Test explosive squib from each shear valve	3.6.1.3.11	-----------
Verify each purge valve is blocked	3.6.1.3.15	------------
Drywell Pressure	**3.6.1.4**	-----------
Verify drywell pressure is within limit	3.6.1.4.1	------------
Drywell Average Air Temperature	**3.6.1.5**	-----------
Verify drywell average air temperature is within limit	3.6.1.5.1	------------
LLS Valves	**3.6.1.6**	-----------
Verify each LLS valve opens when manually actuated	3.6.1.6.1	------------
Verify LLS system actuates on initiation signal	3.6.1.6.2	------------
Reactor Building - Suppression Chamber Vacuum Breakers	**3.6.1.7**	**3.3.6**
Verify each vacuum breaker is closed	3.6.1.7.1	-----------
Perform functional test on each vacuum breaker	3.6.1.7.2	4.3.6.c(1)
Verify opening setpoint for each vacuum breaker	3.6.1.7.3	4.3.6.c(2)
Air-operated vacuum breaker instrumentation calibration	-----------	-----------
Suppression Chamber - Drywell Vacuum Breakers	**3.6.1.8**	**3.3.6**
Verify each vacuum breaker is closed	3.6.1.8.1	-------------
Perform functional test on each vacuum breaker	3.6.1.8.2	4.3.6.a
Verify opening setpoint for each vacuum breaker	3.6.1.8.3	-----------
Vacuum breaker position indication and alarm calibration	-----------	------------
Vacuum breaker inspection	------------	4.3.6.b(3)
Main Steam Isolation Valve (MSIV) Leakage Control System	**3.6.1.9**	**N/A**
Operate each MSIV LCS blower	3.6.1.9.1	-----------
Verify continuity of inboard MSIV LCS heater element	3.6.1.9.2	------------
Perform functional test of each MSIV LCS subsystem	3.6.1.9.3	-----------
Suppression Pool Average Temperature	**3.6.2.1**	**3.3.2**

Technical Specification Section Title/Surveillance Description*	**TSTF-425**	**NMP-1**
Verify suppression pool average temperature within limits	3.6.2.1.1	4.3.2.a
Suppression Pool Water Level	**3.6.2.2**	**3.3.2**
Verify suppression pool water level within limits	3.6.2.2.1	4.3.2.a
Suppression Chamber Level, Temperature and Pressure	----------	**3.3.2**
Suppression chamber temperature and pressure	----------	4.3.2.a
Suppression chamber visual inspection	----------	4.3.2.b
RHR Suppression Pool Cooling	**3.6.2.3**	----------
Verify each valve in flow path is in correct position	3.6.2.3.1	----------
Verify each RHR pump develops flow rate	3.6.2.3.2	----------
RHR Suppression Pool Spray	**3.6.2.4**	----------
Verify each valve in flow path is in correct position	3.6.2.4.1	----------
Verify RHR pump develops flow rate	3.6.2.4.2	4.3.7.a.2
Drywell - Suppression Chamber Differential Pressure	**3.6.2.5**	----------
Verify differential pressure is within limit	3.6.2.5.1	----------
Drywell Cooling System Fans	**3.6.3.1**	----------
Operate each fan ≥ 15 minutes	3.6.3.1.1	----------
Verify each fan flow rate	3.6.3.1.2	----------
Primary Containment Oxygen Concentration	**3.6.3.2**	**3.3.1**
Verify PC oxygen concentration is within limits	3.6.3.2.1	4.3.1
Containment Atmosphere Dilution (CAD) System	**3.6.3.3**	----------
Verify CAD liquid nitrogen storage	3.6.3.3.1	----------
Verify each CAD valve in flow path is in correct position	3.6.3.3.2	----------
Secondary Containment	**3.6.4.1**	**3.4**
Verify SC vacuum is > 0.25 inch of vacuum water gauge	3.6.4.1.1	4.4.1
Verify all SC equipment hatches closed and sealed	3.6.4.1.2	----------
Verify one SC access door in each opening is closed	3.6.4.1.3	4.4.3.b(1)
Verify SC drawn down using one SGTS	3.6.4.1.4	4.4.1
Verify SC can be maintained using one SGTS	3.6.4.1.5	4.4.1
Core and Containment Spray Compartments	----------	4.4.3.a
Secondary Containment Isolation Valves / Rector Building Integrity – Isolation Valves	**3.6.4.2**	**3.4.2**
Verify each SC isolation manual valve is closed	3.6.4.2.1	----------
Verify isolation time of each SCIV	3.6.4.2.2	----------
Verify each automatic SCIV actuates to isolation position	3.6.4.2.3	4.4.2
Standby Gas Treatment (SGT) System	**3.6.4.3**	----------
Operate each SGT subsystem for 10 hours	3.6.4.3.1	----------
SGT filter testing [NUREG-1433 - Ventilation Filter Testing Program]	3.6.4.3.2	----------
Verify each SGT subsystem actuates on initiation signal	3.6.4.3.3	----------
Verify each SGT filter cooler bypass damper can be opened	3.6.4.3.4	----------
Residual Heat Removal Service Water (RHRSW) System	**3.7.1**	----------
Verify each RHRSW valve in flow path in correct position	3.7.1.1	----------
Plant Service Water (PSW) System and Ultimate Heat Sink (UHS)	**3.7.2**	----------
Verify water level in cooling tower basin	3.7.2.1	----------
Verify water level in pump well of pump structure	3.7.2.2	----------
Verify average water temperature of heat sink	3.7.2.3	4.3.7.f

Technical Specification Section Title/Surveillance Description*	TSTF-425	NMP-1
Operate each cooling tower fan	3.7.2.4	------------
Verify each PSW SSW valve in flow path is in correct position	3.7.2.5	------------
Verify PSW SSW actuates on initiation signal	3.7.2.6	------------
Diesel Generator (DG) Standby Service Water (SSW) System	**3.7.3**	------------
Verify each DG SSW valve in flow path is in correct position	3.7.3.1	------------
Verify DG SSW pump starts automatically when DG energizes the respective bus	3.7.3.2	4.6.3.b
MCREC [Control Room HVAC] System / Control Room Air Treatment System	**3.7.4**	------------
Operate each MCREC [Control Room HVAC] subsystem	3.7.4.1	4.4.5.e
Test and Sampling analyses	------------	4.4.5.b
Verify each subsystem actuates on initiation signal	3.7.4.3	4.4.5.f
Verify each subsystem can maintain positive pressure	3.7.4.4	------------
Verify DP across HEPA filters	------------	4.4.5.a
Control Room Air Conditioning System	**3.7.5**	------------
Verify each subsystem has capability to remove heat load	3.7.5.1	------------
Emergency Ventilation System	------------	**3.4.4**
Pressure drop across HEPA and Charcoal adsorbers	------------	4.4.4.a(1)
Test and sample analysis	------------	4.4.4.b
Each circuit shall be operated	------------	4.4.4.e
Automatic initiation of each branch	------------	4.4.4.g
Operability of bypass valve for filter cooling	------------	4.4.4.h
Surveillances with inoperable components	------------	4.4.4.i
Main Condenser Offgas	**3.7.6**	**3.6.15**
Verify gross gamma activity rate of the noble gases	3.7.6.1	4.6.15
Main Turbine Bypass System	**3.7.7**	------------
Verify one complete cycle of each main turbine bypass valve	3.7.7.1	------------
Perform system functional test	3.7.7.2	------------
Verify Turbine Bypass System Response Time within limits	3.7.7.3	------------
Spent Fuel Storage Pool Water Level	**3.7.8**	------------
Verify spent fuel storage pool water level	3.7.8.1	------------
AC Sources - Operating	**3.8.1**	**3.6.3**
Verify correct breaker alignment	3.8.1.1	------------
Verify each DG starts from standby conditions/steady state	3.8.1.2	4.6.3.b
Verify each DG is synchronized and loaded	3.8.1.3	4.6.3.b
Verify each day tank level	3.8.1.4	------------
Check for and remove accumulated water from day tank	3.8.1.5	------------
Verify fuel oil transfer system operates	3.8.1.6	------------
Verify each DG starts from standby conditions/quick start	3.8.1.7	4.6.3.b
Verify transfer of power from offsite circuit to alternate circuit	3.8.1.8	------------
Verify DG rejects load greater than single largest load	3.8.1.9	------------
Verify DG maintains load following load reject	3.8.1.10	------------
Verify on loss of offsite power signal	3.8.1.11	4.6.3.a
Verify DG starts on ECCS initiation signal	3.8.1.12	4.6.3.a
Verify DG automatic trips bypassed on ECCS initiation signal	3.8.1.13	------------
Verify each DG operates for > 24 hours	3.8.1.14	------------

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Verify each DG starts from standby conditions/quick restart	3.8.1.15	-------------
Verify each DG synchronizes with offsite power	3.8.1.16	-------------
Verify ECCS initiation signal overrides test mode	3.8.1.17	-------------
Verify interval between each timed load block	3.8.1.18	-------------
Verify on LOOP in conjunction with ECCS initiation signal	3.8.1.19	4.6.3.a
Verify simultaneous DG starts	3.8.1.20	-------------
Diesel Generator Starting Batteries	-------------	-------------
Verify electrolyte level, pilot cell voltage, battery voltage on float charge, pilot cell specific gravity	-------------	-------------
Specific gravity and electrolyte temperature for each fourth cell	-------------	-------------
Specific gravity and electrolyte temperature for every cell	-------------	-------------
Battery capacity during battery service test	-------------	-------------
Battery capacity during battery discharge test	-------------	-------------
Diesel Fuel Oil, Lube Oil, and Starting Air	**3.8.3**	-------------
Verify fuel oil storage tank volume	3.8.3.1	-------------
Verify lube oil inventory	3.8.3.2	-------------
Verify each DG air start receiver pressure	3.8.3.4	-------------
Check/remove accumulated water from fuel oil storage tank	3.8.3.5	-------------
DC Sources – Operating/Battery Parameters	**3.8.4**	-------------
Verify battery terminal voltage	3.8.4.1	4.6.3.c
Verify each battery charger supplies amperage	3.8.4.2	-------------
Verify battery capacity during battery service test	3.8.4.3	-------------
Battery Parameters	**3.8.6**	-------------
Verify battery float current	3.8.6.1	-------------
Verify battery pilot cell voltage	3.8.6.2	4.6.3.c
Verify battery connected cell electrolyte level	3.8.6.3	-------------
Verify battery pilot cell temperature	3.8.6.4	-------------
Verify battery connected cell voltage	3.8.6.5	-------------
Verify battery capacity during performance discharge test	3.8.6.6	-------------
Inverters - Operating	**3.8.7**	-------------
Verify correct inverter voltage, frequency and alignment	3.8.7.1	-------------
Inverters - Shutdown	**3.8.8**	-------------
Verify correct inverter voltage, frequency and alignment	3.8.8.1	-------------
Distribution System - Operating	**3.8.9**	-------------
Verify correct breaker alignment/power to distribution subsystems	3.8.9.1	-------------
Distribution System - Shutdown	**3.8.10**	-------------
Verify correct breaker alignment/power to distribution subsystems	3.8.10.1	-------------
Refueling Equipment Interlocks	**3.9.1**	**3.5.2**
Channel Functional Test of refueling equip interlock inputs	3.9.1.1	4.5.2
SRMs (Refueling)	-------------	**3.5.1**
SRMs checked, tested, and calibrated	-------------	4.5.1
Refuel Position One-Rod-Out Interlock	**3.9.2**	-------------
Verify reactor mode switch locked in refuel position	3.9.2.1	-------------
Perform Channel Functional Test	3.9.2.2	-------------
Control Rod Position	**3.9.3**	-------------
Verify all control rods fully inserted	3.9.3.1	-------------

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Control Rod Operability - Refuel	**3.9.5**	-------------
Insert each withdrawn control rod one notch	3.9.5.1	-------------
Verify each withdrawn control rod scram accumulator press	3.9.5.2	-------------
Reactor Pressure Vessel (RPV) Water Level - Irradiated Fuel	**3.9.6**	-------------
Verify RPV water level	3.9.6.1	-------------
Reactor Pressure Vessel (RPV) Water Level - New Fuel	**3.9.7**	-------------
Verify RPV water level	3.9.7.1	-------------
RHR - High Water Level	**3.9.8**	-------------
Verify one RHR shutdown cooling subsystem operating	3.9.8.1	-------------
RHR - Low Water Level	**3.9.9**	-------------
Verify one RHR shutdown cooling subsystem operating	3.9.9.1	-------------
Reactor Mode Switch Interlock Testing	**3.10.2**	-------------
Verify all control rods fully inserted in core cells	3.10.2.1	-------------
Verify no core alterations in progress	3.10.2.2	-------------
Single Control Rod Withdrawal - Hot Shutdown	**3.10.3**	-------------
Verify all control rods in five-by-five array are disarmed	3.10.3.2	-------------
Verify all control rods other than withdrawn rod are fully inserted	3.10.3.3	-------------
Single Control Rod Withdrawal - Cold Shutdown	**3.10.4**	-------------
Verify all control rods in five-by-five array are disarmed	3.10.4.2	-------------
Verify all control rods other than withdrawn rod are fully inserted	3.10.4.3	-------------
Verify a control rod withdrawal block is inserted	3.10.4.4	-------------
Single Control Rod Drive (CRD) Removal - Refuel	**3.10.5**	-------------
Verify all control rods other than withdrawn rod are fully inserted	3.10.5.1	-------------
Verify all control rods in five-by-five array are disarmed	3.10.5.2	-------------
Verify a control rod withdrawal block is inserted	3.10.5.3	-------------
Verify no core alterations in progress	3.10.5.5	-------------
Multiple CRD Removal-Refuel	**3.10.6**	-------------
Verify four fuel assemblies removed from core cells	3.10.6.1	-------------
Verify all other rods in core cells inserted	3.10.6.2	-------------
Verify fuel assemblies being loaded comply with reload sequence	3.10.6.3	-------------
Shutdown Discharge Margin Test - Refueling	**3.10.8**	-------------
Verify no other core alterations in progress	3.10.8.4	-------------
Verify CRD charging water header pressure	3.10.8.6	-------------
Recirculation Loops - Testing	**3.10.9**	-------------
Verify LCO 3.4.1 requirements suspended for < 24 hours	3.10.9.1	-------------
Verify Thermal power < 5% RTP during Physics Test	3.10.9.2	-------------
Training Startups	**3.10.10**	-------------
Verify all operable IRM channels are <25/40 div. of full scale	3.10.10.1	-------------
Verify average reactor coolant temperature < 200 F	3.10.10.2	-------------
Radioactive Liquid Storage	-------------	-------------
Liquids contained in tanks sampled and analyzed	-------------	-------------
Programs (Surveillance Frequency Control Program [SFCP])	**5.5.15**	6.5.9
High Pressure Core Injection	-------------	**3.1.8**
Pump auto start	-------------	4.1.8.a
Pump operability	-------------	4.1.8.b
Surveillances with inoperable components	-------------	4.1.8.c

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Radioactive Material Sources	-------------	**3.6.5**
Seal source leak test	-------------	4.6.5.1
Startup sources	-------------	4.6.5.3
Reactor Coolant System Isolation Valves	-------------	**3.27**
Automatic Initiation and Closure Times	-------------	4.2.7.a
Feedwater and Main Steam partial closure and re-opening	-------------	4.2.7.c
Pressure Relief Systems-Safety Valves	-------------	**3.2.8**
Remove and test for lift set point and partial test	-------------	4.2.8
Containment Spray System	-------------	**3.3.7**
Automatic Start of Containment Spray Pump	-------------	4.3.7.a(1)
Pump Operability	3.6.2.4.2	4.3.7.a(2)
Raw Water Cooling Pump Operability	-------------	4.3.7.c
Surveillance with Inoperable equipment	-------------	4.3.7.d
Lake water Temperature	3.7.2.3	4.3.7.f
Instruments that Initiates Core Spray	-------------	**Per Table 4.6.2.d**
Instruments that Initiates Containment Spray	-------------	**Per Table 4.6.2.e**
Instrumentation that Initiates Control Rod Withdrawal Block	-------------	**Per Table 4.6.2.g**
Vacuum Pump Isolation	-------------	**Per Table 4.6.2.h**
Emergency Ventilation Initiation	-------------	**Per Table 4.6.2.j**
High Pressure Coolant Injection	-------------	**Per Table 4.6.2.k**
Control Room Air Treatment System Initiation	-------------	**Per Table 4.6.2.l**
Mechanical Vacuum Pump Isolation	-------------	**3.6.1**
Automatic isolation of the mechanical vacuum pump	-------------	4.6.1.b
Special Test Exception –SDM Demonstration	-------------	**3.7.1**
Operability of SRM and RWM	-------------	4.7.1.a

* The Technical Specification Section Title/Surveillance Description portion of this attachment is a summary description of the referenced TSTF-425 (NUREG-1433)/NMP-1 TS Surveillances which is provided for information purposes only and is not intended to be a verbatim description of the TS Surveillances.

ATTACHMENT 5

License Amendment Request

Nine Mile Point Nuclear Station, Unit 1
Docket No. 50-220

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

Proposed No Significant Hazards Consideration

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 General Electric Plants, BWR/4 (NUREG-1433), to allow relocation of specific 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) related to the Relocation of 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).

Basis for proposed no significant hazards consideration: As required by 10 CFR 50.91(a), the Exelon 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 technical specifications for which the surveillance frequencies are relocated are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

   Therefore, the proposed 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, Exelon 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 above, Exelon 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

License Amendment Request

Nine Mile Point Nuclear Station, Unit 1
Docket No. 50-220

Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)

Proposed Inserts

INSERT 1

In [in] accordance with the Surveillance Frequency Control Program

INSERT 2

6.5.9 Surveillance Frequency Control program

This program provides controls for the Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

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

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

c. The provision of Surveillance Requirement 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

INSERT 3

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