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10 CFR 50.90

June 4, 2015

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

R.E. Ginna Nuclear Power Plant
Renewed Facility Operating License No. DPR-18
NRC Docket No. 50-244

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)

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 Generation Company, LLC (Exelon) is submitting a request for an amendment to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. DPR-18 for R. E. Ginna Nuclear Power Plant (Ginna).

The proposed amendment would modify Ginna's 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, 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.

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. Attachment 3 provides the existing Ginna TS pages marked up to show the proposed changes. Attachment 4 provides the proposed Ginna TS Bases changes. Attachment 5 provides a TSTF-425 versus Ginna TS Cross-Reference. Attachment 6 provides the proposed No Significant Hazards Consideration. Attachment 7 provides the proposed inserts.

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There are no regulatory commitments contained in this letter.

Exelon requests approval of the proposed license amendment by June 4, 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 4th day of June 2015.

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

Respectfully,

David T. Helker 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 Page Changes
4. Proposed Technical Specification Bases Page Changes
5. TSTF-425 (NUREG-1431) vs. Ginna Cross-Reference
6. Proposed No Significant Hazards Consideration
7. Proposed Inserts

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

ATTACHMENT 1

License Amendment Request

R. E. Ginna Nuclear Power Plant
Docket No. 50-244

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 R. E. Ginna Nuclear Power plant (Ginna) Technical Specifications (TS) by relocating specific TS 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 5, 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). Ginna was not licensed to the 10 CFR 50, Appendix A GDC. However, the Ginna's Updated Final Safety Analysis Report (UFSAR), in Section 3.1 "Conformance with NRC General Design Criteria," provides an assessment against the GDC. Based on the assessment performed and described in the in the Ginna UFSAR, Exelon believes that the plant-specific requirements for Ginna 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 Ginna and justify this amendment to incorporate the changes to the Ginna 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 Ginna 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

After NRC approval of TSTF-425, it was recognized that surveillance frequencies that have not been changed under the Surveillance Frequency Control Program (SFCP) may not be based on operating experience, equipment reliability or plant risk. Therefore, the TSTF and the NRC agreed that the TSTF-425 TS Bases insert, "The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program," should be revised to state, "The Surveillance Frequency is controlled under the Surveillance Frequency Control Program." The existing TS Bases information will be relocated to the licensee-controlled SFCP.

Attachment 5 provides a cross-reference between TSTF-425 versus the Ginna Surveillances included in this amendment request. Attachment 5 includes a summary description of the referenced TSTF-425 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. Surveillances included in TSTF-425 and corresponding Ginna Surveillances have differing Surveillances numbers,
b. Surveillances included in TSTF-425 that are not contained in the Ginna TS, and
c. Ginna plant-specific Surveillances that are not contained in TSTF-425 Surveillances and, therefore, are not included in the TSTF-425 mark-ups.

In addition, there are Surveillances contained in TSTF-425 that are not contained in the Ginna TS. Therefore, the NUREG-1431 mark-ups included in TSTF-425 for these Surveillances are not applicable to Ginna. 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).

Ginna TS include plant-specific Surveillances that are not contained in NUREG-1431 and, therefore, are not included in the NUREG-1431 mark-ups provided in TSTF-425. Exelon has determined that the relocation of the Frequencies for these Ginna plant-specific 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 Surveillances would be controlled under the SFCP. The SFCP provides the necessary administrative controls to require that Surveillances related to testing, calibration and inspection are conducted at a frequency to assure 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 Ginna plant-specific 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 Ginna, and is provided as Attachment 6 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 Ginna.

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 but not limited to Amendment Nos. 186 and 147 for Limerick Generating Station, Amendment No.276 for Oyster Creek Nuclear Power Station dated September 27, 2010; Amendment Nos. 200 and 201 for Diablo Canyon Power Plant, Units 1 and 2, respectively, dated October 30, 2008; and Amendment Nos. 188 and 175 for South Texas Project, Units 1 and 2, respectively, dated October 31, 2008. The subject License Amendment Request proposes to relocate periodic surveillance frequencies to a licensee-controlled program and add a new program (the Surveillance Frequency Control Program) to the Administrative Controls section of TS in accordance with TSTF-425 and as discussed in the previously approved amendments.

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 Ginna, 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

R. E. Ginna Nuclear Power Plant
Docket No. 50-244

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

1.0 Overview

The implementation of the Surveillance Frequency Control Program (also referred to as Technical Specifications Initiative 5b) at the Ginna Nuclear Power Plant will follow the guidance provided in NEI 04-10, Revision 1 [Ref. 1] in evaluating proposed surveillance test interval (STI; also referred to as "surveillance frequency") changes.

The following steps of the risk-informed STI revision process are common to 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 Decision making Panel (IDP), which is normally the same panel as is used for Maintenance Rule implementation, but with the addition of specialists with experience in surveillance tests and system or component reliability. If the IDP approves the STI revision, the change is documented and implemented, and available for audit by the Nuclear Regulatory Commission (NRC). If the IDP does not approve the STI revision, the STI value is left unchanged.
- Performance monitoring is conducted as recommended by the IDP. In some cases, no additional monitoring may be necessary beyond that already conducted under the Maintenance Rule. The performance monitoring helps to confirm that no failure mechanisms related to the revised test interval become important enough to alter the information provided for the justification of the interval changes.
- The IDP is responsible for periodic review of performance monitoring results. If it is determined that the time interval between successive performances of a surveillance test is a factor in the unsatisfactory performances of the surveillance, the IDP returns the STI 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 test interval 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 interval requests. Item 3 satisfies one of the requirements of Section 4.2 of RG 1.200. The remaining requirements of Section 4.2 are addressed by Item 4 below.):

1. Identify the parts of the PRA used to support the application
   - 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 Regulatory Guide (RG-1.200 Revision 1 was used for the Ginna Internal Events PRA Peer review). 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.

2.0 Technical Adequacy of the PRA Model

The GN114A-W version of the Ginna PRA model is the most recent evaluation of internal event risks. The Ginna 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 Ginna 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 Ginna 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 Ginna PRA.

2.0.1 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 T&RM 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 an update of the Mitigating System Performance Indicator (MSPI) application. Exelon will be performing a Full Power Internal Events (FPIE) model update to the Ginna PRA in 2015. As indicated previously, RG 1.200 also requires that additional information be provided as part of the 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.

2.0.2 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 GINNA will be performed and an assessment of the impact on the results of the application will be made prior to presenting the results of the risk analysis to the IDP. If a non-trivial impact is expected, then this may include the performance of additional sensitivity studies or PRA model changes to confirm the impact on the risk analysis.

2.0.3 Applicability of Peer Review Findings and Observations

A PRA model update was completed in 2009, resulting in the Ginna PRA Model 6.5. The Ginna PRA model was revised to meet RG 1.200, revision 1, guidance and comply with the ASME/ANS PRA Standard RA-Sc-2007[Ref. 3].

This model was peer reviewed under the auspices of the PWR Owners Group (PWROG) in the 2nd quarter of 2009 [Ref. 7]. This peer review was performed following NEI 05-04 [Ref. 5], and NEI 00-002 [Ref. 6]. 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 2012 to support implementation of NFPA-805. As part of the development of this model a peer review of the fire PRA was conducted in June of 2012 [Ref. 8]. This peer review used NEI-07-12 [Ref. 9] to evaluate the model against the ASME PRA Standard (ASME/ANS RA-Sa-2009) [Ref. 10] along with the NRC clarifications provided in Regulatory Guide 1.200, Rev. 2 [Ref. 22].

Since the 2012 peer review, several updates to the Ginna PRA have taken place. An application specific model update was completed in December 2014 to support the Mitigating System Performance Indicator (MSPI) process. The latest model is the GN114A-W. This model includes the addition of two diesel generators for providing an alternate source of power to the Standby AFW (SAFW) Pumps and a condensate storage tank to provide a dedicated source of water to the SAFW Pumps.

2.0.4 Consistency with Applicable PRA Standards

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

- The 2009 peer review for the PRA ASME model update identified 309 Supporting Requirements (SR) applicable to the Ginna 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 cc 3, and 198 fully met all capability requirements. There were 24 findings and observations (F&O's) issued to address the identified gaps to compliance with the PRA standard. Subsequent to the peer review, 13 of the findings have been addressed and 11 are still open pending the next model update. The F&O's are listed in Table 2-1 which includes what, if any impact, there may be to the assessment of STIs for the 5b initiative.
- The 2012 fire PRA peer review for the PRA ASME model update identified 183 Supporting Requirements (SR) to be reviewed for the Ginna PRA. Of these 2 were not met, 2 met capability category (cc) 1, 8 partially met cc 2, 17 met cc 2, 13 partially met cc 3, 7 met cc 3, and 118 fully met all capability requirements and 16 were not applicable. There were 19 findings and 22 suggestions issued to address potential gaps to compliance with the PRA standard. There were 3 Best Practices. All of the findings from the fire PRA peer review have since been closed. As the results of this peer review have already been communicated to the NRC as part of the NFPA-805 submittal [Ref. 12] and subsequent requests for additional information (RAI), 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 a non-trivial impact is expected, then this may include the performance of additional sensitivity studies or PRA model changes to confirm the impact on the risk analysis.

2.0.5 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 for each STI change assessment will be documented and included in the results of the risk analysis that goes to the IDP.

2.1 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 GINNA Individual Plant Examination of External Events (IPEEE) submittal in response to the NRC IPEEE Program (Generic Letter 88-20, Supplement 4) [Ref. 4]. 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 primary areas of external event evaluation at GINNA were internal fires and seismic risk. The internal fire events were addressed by using a combination of the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology [Ref. 14] and fire PSA. The results of the Fire Analysis are documented in the R.E. Ginna Nuclear Power Plant IPEEE Fire Analysis transmitted to the NRC in June 1998 [Ref. 13]. The seismic evaluations were performed in accordance with Generic Implementation Procedure (GIP) developed by the Seismic Qualification Utility Group (SQUG) of which Ginna was a member. The GIP provided plants a method for addressing Unresolved Safety Issue A-46 (Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors (USI A-46). Beyond this, Ginna performed a reduced-scope IPEEE for seismic events to close out IPEEE for Seismic Events. The Ginna USI A-46 Seismic Evaluation Report and the IPEEE Seismic Evaluation Report were transmitted to the NRC in January 1997 [Ref. 15] However, there are no comprehensive CDF and LERF values available from the seismic IPEEE report to support the STI risk assessments.

High Winds, External Floods and Transportation Accidents were reviewed against the Standard Review Plan (SRP) as Ginna was one of the eleven participants in the NRC's Systematic Evaluation Program (SEP). Following plant modifications, it was determined that the Ginna plant met the Standard Review Plan criteria. Based on the NRC Safety Evaluation Reports (SERs) for Ginna's SEP results, no further submittals for GL 88-20 Supplement 4 were warranted for high winds, external floods, or transportation accidents.

Since the performance of the IPEEE, Ginna has submitted a License Amendment Request for conversion from appendix R compliance to NFPA-805 for fire protection [Ref. 12]. Pursuant to this change, a fire PRA has been created and implemented at GINNA. This Fire PRA model was created under the auspices of NUREG/CR-6850 [Ref. 16] and has undergone PWROG peer review (completed August 2012) [Ref. 11]. The Ginna Fire PRA was developed using the National Institute of Standards and Technology (NIST) Consolidated Model of Fire and Smoke Transport (CFAST) Methodology [Ref. 18]; the Fire Dynamics Simulator, also developed by NIST; NUREG-1805 Fire Dynamics Tools (FDTs) computational Spreadsheets [Ref. 21]; EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [Ref. 16] and the associated NUREG/CR-6850 Frequently Asked Questions (FAQ) Process [Ref. 20]; Fire Events Database [Ref. 19] and plant specific data. This fire PRA has numerous capabilities not considered in the IPEEE fire PRA model including explicit analysis of all risk significant fire areas such as the main control room (MCR) and Relay Room (RR). Multiple spurious operations (MSO) considerations are also included. The ignition frequencies for all fire areas were developed using the guidance in NUREG/CR-6850 [Ref. 16] and also incorporate revised guidance for ignition frequencies [Ref. 17].

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.

2.2 Summary

The GINNA PRA technical capability evaluations and the maintenance and update processes described above provide a robust basis for concluding that the GINNA 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.

2.3 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] 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.

[4] 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.

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

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

[7] RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements for R. E. Ginna Station Probabilistic Risk Assessment, Project PA-RMSC-0386, August 2009.

[8] Fire PRA Peer Review Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard for L1/LERF PRA for NPP Applications for the Ginna Station Fire PRA, Project PA-RMSC-0403, August 2012.

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

[10] 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.

[11] Fire PRA Peer Review Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications for the Ginna Station Fire Probabilistic Assessment, August 2012.

[12] Letter from Mr. Joseph E. Pacher (Ginna LLC) to Document Control Desk (NRC), dated March 28, 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 (ADAMS Accession No. ML 13093A064).

[13] Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), 1. Ginna Station Fire IPEEE, RE Ginna Nuclear Power Plant; 2 Hydrogen Storage Facility, dated June 30, 1998.

[14] Fire-Induced Vulnerability Evaluation (FIVE) Methodology Plant Screening Guide, EPRI TR-100370, Electric Power Research Institute, Final Report, April 1992.

[15] Letter from RC Mecredy (RG&E) to Guy S. Vissing (NRC), Resolution of GL 87-02, Supplement 1 and GL 88-20, Supplements 4 and 5 (Seismic Event Only) RG&E Corp, R.E. Ginna Nuclear Power Plant, dated January 31, 1997.

[16] EPRI/NRC-RES, Fire PRA Methodology for Nuclear Power Facilities, EPRI 1011989, NUREG/CR-6850, Final Report, September 2005.

[17] Fire Probabilistic Risk Assessment Methods Enhancements Supplement 1 to NUREG/CR-6850 and EPRI 1011989, EPRI 1019259, Electric Power Research Institute, December 2009.

[18] National Institute of Standards and Technology's (NIST) Consolidated Model of Fire Growth and Smoke Transport (CFAST) Version (6) (Jones et al., 2004)

[19] NSAC/179L, Electric Power Research Institute, Fire Events Database for U.S. Nuclear Power Plants, Rev. 1, January, 1993.

[20] Letter from John A. Grobe (NRR) to Alexander Marion (NEI), dated June 1, 2009, Path Forward in Resolving Frequently Asked Questions Related to NUREG/CR-6850. Accession No. ML090920045.

[21] NUREG-1805, Supplement 1, Volumes 1 & 2, Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program.

[22] Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities, Revision 2, March, 2009

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IE-C12 [2005: IE-C10] URE 845	COMPARE results and EXPLAIN differences in the initiating event analysis with generic data sources to provide a reasonableness check of the results.	Open	**F&O IE-C10-01:** The Ginna Initiating Event Notebook (G1-IE-0001, Rev. 1) Section 4.3 provides a cross-reference between the Ginna Initiating Events and the "NRC Rates of Initiating Events" in table 4-7. Table 4-7 cross-reference includes columns for NUREG/CR-5750 Category and NP-2230 EPRI/NUREG/CR-3862 PWR Category. Table 4-8 provides a cross-reference between Ginna and similar PWR plants (Point Beach, Prairie Island, and Kewaunee). An explanation of differences in Initiating Events between Ginna and similar PWRs is contained in the PRA Quantification (QU) Notebook (G1-QU-0001, Rev. 0) Table 4-5 "Comparison of Ginna Core Damage Results to Similar Plants". However, no explanation of differences between plant-specific initiating events and generic initiating events was located in either the Initiating Event Notebook (G1-IE-0001, Rev. 1) or QU Notebook (G1-QU-0001, Rev. 0).	Documentation only: Provide comparison of core damage results based on generic data cross-referenced in Table 4-7.	This item is a documentation issue. No impact on TSTF-425 analysis.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IE-C15 [2005: IE-C13]	CHARACTERIZE the uncertainty in the initiating event frequencies and PROVIDE *mean values for use in the* quantification of the PRA results.	Open	**F&O IE-C13-01:** G1-IE-0001, PRA INITIATING EVENT (IE) NOTEBOOK, Section 5 *documents assumptions and sources of* uncertainty. However, section 5 does not provide or reference the parametric uncertainty initiating event data distribution. For example, the distribution for TIGRLOSP is identified in the CAFTA model, newauto_65a-w-Fld.caf, has having an EF of 7.39. However, no documentation for the error factor could be found. Therefore, this SR is not met.	Documentation only: Include error factors and brief discussion about IE *frequency uncertainty.*	This item is a documentation issue and IE frequency *distribution evaluation.* Changes will not impact the TSTF-425 analysis

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
SC-A2	SPECIFY the plant parameters (e.g., highest node temperature, core collapsed liquid level) and associated acceptance criteria (e.g., temperature limit) to be used in determining core damage. Select these parameters such that determination of core damage is as realistic as practical, in a manner - consistent with current best practice. DEFINE computer code-predicted acceptance criteria with sufficient margin on the code-calculated values to allow for limitations of the code, sophistication of the models, and uncertainties in the results, in a manner consistent with the requirements specified under HLR-SC-B. Examples of measures for core damage suitable for Capability Category II/III, that have been used in PRAs, include (a) collapsed liquid level less than 1?3 core height or code-predicted peak core temperature >2,500°F (BWR) (b) collapsed liquid level below top of active fuel for a prolonged period, or code-pre-dicted core peak node temperature >2,200°F using a code with detailed core modeling; or code-predicted core peak node temperature >1,800°F using a code with simplified (e.g., single-node core model, lumped para-meter) core modeling; or code-predicted core exit temperature >1,200°F for 30 min using a code with simplified core modeling (PWR).	Open	**F&O SC-A2-01:** The definition of core damage documented in the Ginna-AS-Notebook-Rev-1 Section 2.2 is consistent with the examples of measures for core damage suitable for Capability Category I as defined in NUREG/CR-4550. For Category II Ginna could use the code-predicted core exit temperature >1,200°F for 30 min using PCTRAN (code with simplified core modeling (PWR)).	We agree with the peer reviewers that the approach taken in the Ginna PRA is overly conservative and not consistent with the requirements of Category II. The peer reviewers suggested using a core exit temperature of 1200°F for 30 minutes as the criterion for core damage, but we would recommend using either that criterion or a peak core node temperature of 1800°F. Based on a review of the PCTRAN results, it is likely that the 1800°F peak core temperature would be reached earlier than the time at which the core exit temperature would be greater than 1200°F for 30 minutes.	Over the typical complete loss of decay heat removal timing success criteria, the delta time between core uncovery and CET temperatures reach 1200°F for 30 minutes or 1800° peak center line is fairly small. As such, the timing benefit is not expected to be large except for the fast moving events such as large break LOCAs. For these events, we use the UFSAR success criteria. Although this is not expected to be a significant effect, we do remain a conservative CAT I. Therefore, the model used for TSTS-425 analysis may be conservative.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
SC-A4	IDENTIFY mitigating systems that are shared between units, and the manner in which the sharing is performed should both units experience a common initiating event (e.g., LOOP).	Complete	F&O SC-A4-02 - Operator action RCHFDX1BAF (operator fails to align BAF given 1 of 2 PORVs and no charging) is not included in the fault tree model. It appears that this event should be added in Event Tree TIU Sequence 5 Failures under gate TL_FB. This is an omission in the model and may affect CDF and LERF.	Add RCHFDX1BAF to the Event Tree TIU, as appropriate.	No impact to TSTF 425. Action placed in Event Tree TIU logic and Finding addressed.
SY-A10 [SY-A11 – 2005]	INCORPORATE the effect of variable success criteria (i.e., success criteria that change as a function of plant status) into the system modeling. *Example causes of variable system* success criteria are *(a) different accident scenarios.* Different success criteria are required for some systems to mitigate different accident scenarios (e.g., the number of pumps required to operate in some systems is dependent upon the modeled initiating event). *(b) dependence on other components.* Success criteria for some systems are also dependent on the success of another component in the system (e.g., operation of additional pumps in some cooling water systems is required if noncritical loads are not isolated). *(c) time dependence.* Success criteria for some systems are time-dependent (e.g., two pumps are required to provide the needed flow early following an accident initiator, but only one is required for mitigation later following the accident). *(d) sharing of a system between units.* Success criteria may be affected when both units are challenged by the same initiating event (e.g., LOOP).	Complete	SY-A11-01 - Gate TL_FBHRD1 input to gate TL_FB for failure of Bleed and Feed models success as requiring 1 SI pump and 1 PORV. The logic does not include 75 gpm charging flow which is noted in the Success Criteria notebook as required to support single PORV success. This was confirmed through discussion with Ginna PRA personnel. The omission of a needed mitigating system for support of the Bleed and Feed function may underestimate the importance of these sequences for applications.	Review the Bleed and Feed modeling to ensure the system failures appropriately reflect the success criteria.	No impact as the Finding has been addressed and the logic has been updated and documented in the Success Criteria Notebook.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
SY-A14 [SY-A13 2005]	When identifying the failures in SY-A11 INCLUDE consideration of all failure modes, consistent with available data and model level of detail, except where excluded using the criteria in SY-A15. For example, *(a)* active component fails to start *(b)* active component fails to continue to run *(c)* failure of a closed component to open *(d)* failure of a closed component to remain closed *(e)* failure of an open component to close *(f)* failure of an open component to remain open *(g)* active component spurious operation *(h)* plugging of an active or passive component *(i)* leakage of an active or passive component *(j)* rupture of an active or passive component *(k)* internal leakage of a component *(l)* internal rupture of a component *(m)* failure to provide signal/operate (e.g., instrumentation) *(n)* spurious signal/operation *(o)* pre-initiator human failure events (see SY-A16) *(p)* other failures of a component to perform its required function 62	Complete	SY-A13-02 - Inconsistencies existed in the system modeling of the city water system. Where used to support the GE-Betz system, a basic event for unavailability of city water due to grid LOOP was added (basic event CDAACITYWATER). This same event was not added to the city water modeling for support of the SAFW system.	Review the need to add the unavailability event in the SAFW System.	No impact to TSTF 425. The dependencies for SAFW have been updated in the Ginna PRA.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
SY-A19 [SY-A18 2005]	In the systems model, INCLUDE out-of-service unavailability for components in the system model, unless screened, in a manner consistent with the actual practices and history of the plant for removing equipment from service. *(a)* INCLUDE *(1)* unavailability caused by testing when a component or system train is reconfigured from its required accident mitigating position such that the component cannot function as required *(2)* maintenance events at the train level when procedures require isolating the entire train for maintenance *(3)* maintenance events at a sub-train level (i.e., between tagout boundaries, such as a functional equipment group) when directed by procedures *(b)* Examples of out-of-service unavailability to be modeled are as follows: *(1)* train outages during a work window for preventive/corrective maintenance *(2)* a functional equipment group (FEG) removed from service for preventive/corrective maintenance *(3)* a relief valve taken out of service	Open	SY-A18-01 - Ginna PRA System Notebooks provides a list of all the modeled T&M terms in Section 3.4.C. Section 2.9 of the notebooks provide discussion of procedures and testing that result in Unavailability. The review of these sections found no instances of simultaneous unavailability that can result from planned activities. However, the PRA engineer noted in a discussion that some systems are shadowed in planned maintenance. There is not a specific discussion on plant maintenance practices within the (a)(4) program that would result in planned unavailability of multiple systems OOS (i.e., EDG outages combined with AFW motor driven pump outages to lower total risk as opposed to performing the work independently), or of planned activities resulting in multiple components OOS that do not violate technical specifications (e.g., two AFW pumps in maintenance or an AFW and SAFW pump in maintenance at the same time). If work is done in this manner, it may be appropriate to account for the unavailability of both SSCs in a single term. Modeling of station maintenance practices that result in planned maintenance evolutions with more than a single PRA component OOS (i.e., shadowing equipment outages) can help to minimize the number of random failure sequences and ensure there is not "double counting" of unavailability in the PRA.	Determine if any maintenance practices are performed that result in overlapping unavailability of multiple systems. If it is determined that simultaneous unavailability is possible, model these occurrences as a single unavailability event in the PRA or justify why the unavailability is treated as separate events and include this as a potential source of model uncertainty. Also, consider adding a specific question to the system engineers' questionnaire for each system to determine if there are planned evolutions that represent simultaneous unavailability of multiple SSCs.	If shadowed unavailability is in-fact significantly affecting the unavailability numbers, then this would conservatively affect TSTF-425 analysis. The most significant unavailabilities are related to MSPI related functions which are less likely to include conservative data.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
HR-G3	When estimating HEPs EVALUATE the impact of the following plant-specific and scenario-specific performance shaping factors: (a) quality [type (classroom or simulator) and frequency] of the operator training or experience (b) quality of the written procedures and administrative controls (c) availability of instrumentation needed to take corrective actions (d) degree of clarity of the cues/indications (e) human-machine interface (f) time available and time required to complete the response (g) complexity of the required response (h) environment (e.g., lighting, heat, radiation) under which the operator is working (i) accessibility of the equipment requiring manipulation (j) necessity, adequacy, and availability of special tools, parts, clothing, etc.	Complete	**F&O HR-G3-01:** Details regarding certain elements of the analysis were lacking in the HRA Calculator for a sufficient number of HFEs to judge that this requirement was not met. Evidence that the relevant aspects cited in the SR are addressed for each HFE is critical to assuring that an appropriate analysis has been performed. This is particularly important in the case of HRA, for which the methods are less straightforward than they are for many other parts of the PRA.	Issue: In item (d) of CC II, III, clarify that 'clarity' refers to the meaning of the cues, etc. In item (g) of CC II, III, clarify that complexity refers to both determining the need for and executing the required response. Resolution: Cat I, II, and III: (d) degree of clarity of the meaning of cues / indications (g) complexity of detection, diagnosis and decision-making, and executing the required response.	No impact to TSTF 425. The HRAs have been reviewed to ensure the needed parameters for the evaluation have been populated. CBDM is now used as a max function of CBDT and HCR/ORE. RCHFDMAKEUP as a specific example has a timing basis from Key Input 51. When the annunciator model is used, there is a clear discussion as to the applicability.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
HR-I1	DOCUMENT the human reliability analysis in a manner that facilitates PRA applications, upgrades, and peer review.	Complete	**F&O HR-I1-01:** The bulk of the documentation for the HRA is provided in the HRA Calculator. There are numerous areas in which the documentation is incomplete. The documentation should include a fuller discussion of the cues, bases for timing, specific procedure steps, and other aspects that could affect the analyses.	Documentation only. Same issue as for HR-G3.	No impact to TSTF 425. This item has been addressed. See HR-G3.
QU-B5	Fault tree linking and some other modeling approaches may result in circular logic that must be broken before the model is solved. BREAK the circular logic appropriately. Guidance for breaking logic loops is provided in NUREG/CR-2728 [2-13]. When resolving circular logic, DO NOT introduce unnecessary conservatisms or non-conservatisms.	Open	**F&O QU-B5-01:** In Section 3.1 of the QU Notebook, a mention is made that circular logic checks were performed on the integrated top logic model to ensure it did not exist. An example is listed, but there is no further discussion. System notebooks reviewed generally state in Section 3.3 what was done when circular logic was identified, but no discussion of the methodology was provided nor how conservatisms or non-conservatisms are avoided. No evidence that the required analysis was not performed.	Documentation only: Provide a discussion in the Quantification Notebook Section 3.1 of the methodology used to address circular logic.	The circular logic process is self-revealing when a support gate is added to the tree the CAFTA software identifies a circular logic issue. The circular logic is broken by inserting as much of the logic clip into the tree as possible. Providing more examples of this in the documentation is not expected to affect the TSTF-425 evaluation.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
LE-C2 [2005: LE-C2a]	INCLUDE realistic treatment of feasible operator actions following the onset of core damage consistent with applicable procedures, e.g., EOPs/SAMGs, proceduralized actions, or Technical Support Center guidance.	Open	**F&O LE-C2a-01:** It is conservative to NOT take credit for operator actions post core damage. This is a requirement of the standard to move from Category I to Category II.	There are limited operator actions that could influence LERF at Ginna, so the effect of such actions is not likely to be significant. Moreover, it is likely that there will not be a need for a Category II rating in this area to meet the requirements for most risk-informed applications. One approach to reaching Category II would be to include post-core damage operator actions in the PRA. It is also possible that simply identifying operator actions and showing quantitatively that they will have a negligible impact on LERF will be sufficient to meet the requirements of Category II.	There are limited operator actions that could influence LERF at Ginna, so the effect of such actions is not likely to be significant. If post-core-damage operator actions are credited, LERF estimates could be reduced, but the impact would be minimal. The omission of these operator actions is conservative and does not adversely impact the use of the model for TSTF-425 analysis.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
LE-C11 [2005: LE-C9a]	JUSTIFY any credit given for equipment survivability or human actions that could be impacted by containment failure.	N/A	**F&O LE-C9a-01:** It does not appear that credit was taken for continued operation of equipment and operator actions that could be impacted by containment failure. This is *a requirement of the standard to move* from Category I to Category II.	The requirement is to justify credit taken for equipment survivability or human actions that could be affected by containment failure. Since no such *credit was taken, the SR should have* been judged as not applicable (N/A). This is analogous to the assessment of LE-C7 (old LE-C6) which was judged by the peer reviewers as N/A because human actions that support the accident progression analysis were not credited. Also, note that, in the Calvert Cliffs peer review, the peer reviewers judged this SR as N/A for the same reason. Only if post-containment failure equipment operations or human actions are modeled in the future would it be necessary to provide engineering analysis and written justification as part of the PRA documentation. Otherwise, no additional work is needed.	As no equipment or HRA is credited post-containment failure, the PRA model remains a conservative CAT I.
LE-C13 [2005 LE-C10]	PERFORM a containment bypass analysis in a realistic manner. JUSTIFY any credit taken for scrubbing (i.e., provide an engineering basis for the decontamination factor used).	N/A	**F&O LE-C10-01:** Credit for scrubbing was not taken. A sensitivity for impact of scrubbing was performed and it was determined that the impact of not considering scrubbing is negligible. This is a requirement of the standard to move from Category I to Category II.	Review the possible credit for release scrubbing to reduce LERF.	No impact to TSTF 425. A sensitivity for impact of scrubbing was performed and it was determined that the impact of not considering scrubbing is negligible.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
MU-D1	A PRA Configuration Control Program shall be in place. It shall contain the following key elements: (a) a process for monitoring PRA inputs and collecting new information (b) a process that maintains and upgrades the PRA to be consistent with the as-built, as operated plant (c) a process that ensures that the cumulative impact of pending changes is considered when applying the PRA (d) a process that maintains configuration control of computer codes used to support PRA quantification (e) documentation of the Program	complete	**F&O MU-D1-01** - PRA Configuration Control procedure (GNG-CM-1.01-3003) Step 5.13 provides guidance for updating risk-informed applications. The process described depends upon a database maintained by the Fleet PRA Services Supervisor to identify current living applications requiring change assessment other than those related to maintenance rule performance criteria. No such database could be identified for Ginna. Without a current list of risk-informed applications, the maintenance and update process is dependent upon the knowledge and experience of the staff to know which applications require update. This creates the possibility that an application could be missed in the update process.	The CRMP database has a placeholder for a listing of PRA applications. This portion of the database has been populated to ensure all applications requiring update following a model revision can be easily identified.	This configuration control issue has been addressed. No impact to TSTF 425.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IFSO-A4 [2005: IF-B2]	For each potential source of flooding water, IDENTIFY the flooding mechanisms that would result in a fluid release. INCLUDE: (a) failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc. (b) human-induced mechanisms that could lead to overfilling tanks, diversion of flow through openings created to perform maintenance; inadvertent actuation of fire suppression system (c) other events releasing water into the area	complete	**F&O IF-B2-01:** Failure mechanisms are addressed in conjunction with the calculation of flood frequencies, in Section 5.2 of document 51-9100978-000. Failures of components in piping systems other than tanks are explicitly addressed by the EPRI pipe failure data base. This was the source employed to characterize the frequencies of floods for Ginna. There has, however, been a very limited attempt to address human-induced flood mechanisms, as required by item (b) of SR IF-B2. Such events have been important causes of flooding in the operating experience for US nuclear power plants, and as noted above the assessment of such floods is explicitly required. A more systematic consideration should be made of human-caused floods. This will need to include an assessment of generic data related to human-caused floods, per SR IF-D6.	Address the potential for human-caused flooding in the Internal Flooding Study (51 - 9100978 - 000). Describe the situations where a human error could result in flooding (e.g., inadvertent valve opening, inadvertent train realignment, doors left open) and estimate the probabilities of such events. Model such floods that cannot be screened. Consistent with the Standard, utilize generic data as required by SR IFEV-A7 (IF-D6 in 2005 Standard)	No impact to TSTF 425. Discussion of human caused floods is discussed in detail in Section 3.3 and 5.3 of Internal Flood Notebook (G1-IF-0000-r1) for various systems. Based on the analyses performed, one maintenance induced flood was added to the model, FL-ABO-M-SW – 2,000 gpm SW flood in the Aux Building due to maintenance, isolated within 65 minutes.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IFSN-A6 [2005: IF-C3]	For the SSCs identified in IFSN-A5 (2005 text: IF-C2c), IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. EITHER: a) ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions; OR b) NOTE that these mechanisms are not included in the scope of the evaluation.	Open	**F&O IF-C3-01:** There is no discussion of failures due to jet impingement or pipe whip. There is limited consideration of failure due to humidity/high temperature due to failure of heating steam lines. There is also no discussion of criteria employed to consider the potential for spray failures. To meet Capability Category II, it is necessary either to provide at least a qualitative assessment of the potential for jet impingement and pipe whip, or to state that these failure mechanisms were not considered. It is also required that potential spray failures be evaluated. While spray failures are discussed, there are no criteria specified that would provide assurance that they had been considered in a consistent and adequately comprehensive manner. Provide the requisite discussion of pipe whip and jet impingement to satisfy the standard. Specify appropriate criteria for spray impacts, and assure that the potential spray failures adequately reflect these criteria.	Cat II: INCLUDE failure by submergence and spray in the identification process. ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions. [SAIC note: these mechanisms include submergence, spray, jet impingement, pipe whip, humidity, condensation, temperature concerns] Revise the Internal Flooding Study (51 - 9100978 - 000) to describe the criteria used to determine the potential for failure resulting from spray. Reference Appendix C for a listing of components impacted by spray. Describe how potential spray impact was addressed in the model. Confirm that the assignment of spray impact is consistent with the criteria used. In addition, include a qualitative discussion of the potential impact of jet impingement, pipe whip, humidity, condensation, and temperature effects.	Failures due to jet impingement and pipe whip are now discussed in Section 3.3.1 of the Internal Flood Notebook G1-IF-0000 r1. Failures due to Spray are discussed in Section 3.3.2. Impacts due to spray were assumed to exist within 10 feet of a break location. Spray events are discussed in the IF Flood notebook Section 4.5. Two locations were identified in the Aux Building where Fire Service Water could impact safety related busses and these are explicitly modeled (FL-ABM-FSW-BUS15 and FL-ABO-FSW-BUS14). URE 1179 documents that IF Notebook needs Appendix C completed to complete documentation of spray impacts and modeling of additional spray floods if appropriate. This would be evaluated for any potential impacts to a surveillance frequency interval extension at the time of the evaluation but is not expected to have a significant impact.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IFSN-A8 [2005: IF-C3b]	IDENTIFY inter-area propagation through the normal flow path from one area to another via drain lines; and areas connected via back flow through drain lines involving failed check valves, pipe and cable penetrations (including cable trays), doors, stairwells, hatchways, and HVAC ducts. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads.	complete	**F&O IF-C3b-01:** The analysis does not document consideration of potential barrier failures due to flooding loads (structural failures, failures of doors, etc.) This is required to meet capability categories beyond Capability Category I. Review flood barriers and identify and evaluate any whose failures could contribute adversely to propagation of flooding	Cat II, III: IDENTIFY inter-area. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads and the potential for barrier unavailability, including maintenance activities. Include a discussion of the potential for barrier failure due to flooding, including structures and doors. For walls, a qualitative discussion would appear to be acceptable. For doors, however, specific failure criteria should be developed and described. Flood scenarios should be reviewed and revised, if necessary, to address the potential failure of doors.	A discussion of structural failure of barriers credited as barriers has been added to the IF Notebook r1, Section 4.2.1.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IFSN-A16 [2005: IF-C8]	USE potential human mitigative actions as additional criteria for screening out flood sources if all the following can be shown: (a) flood indication is available in the control room; (b) the flood source can be isolated; and (c) the mitigative action can be *performed with high reliability for the* worst flooding initiator (2005 text: flood from that source). High reliability is established by demonstrating, for example, that the actions are procedurally directed, that adequate time is available for response, that the area is accessible, and that there is sufficient manpower available to perform the actions.	Open	**F&O IF-C8-01:** Only one flood appears to have been screened based on qualitative consideration of potential human action; for that action (2000 gpm FSW break in IBN), there doesn't appear to be any justification for the time identified (190 min). Nothing other than time available is cited as rationale for screening the event. To meet Capability Category II, it is necessary to characterize potential human actions that could terminate flooding more *explicitly than was done in this case.* Address the required aspects for this and any other human actions used in justifying screening out flood scenarios.	Characterize in greater detail those potential human actions that could terminate the event and develop an estimate of the likelihood of failing to mitigate the pipe break using accepted HRA methods.	The screened flood will be added to the flood model (URE 1176). However, the impact is expected to be minimal, and is not expected to have any impact on the SFCP.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IFEV-A6 [2005: IF-D5a]	GATHER plant-specific information on plant design, operating practices, and conditions that may impact flood likelihood (i.e., material condition of fluid systems, experience with water hammer, and maintenance-induced floods). In determining the flood-initiating event frequencies for flood scenario groups, USE a combination of the following (2005 text does not include "of the following") (a) generic and plant-specific operating experience; (b) pipe, component, and tank rupture failure rates from generic data sources and plant-specific experience; (2005 text: and) (c) engineering judgment for consideration of the plant-specific information collected.	Open	**F&O IF-D5a-01:** The current analysis does not adequately address plant-specific characteristics that might affect the manner in which the frequencies of flooding are estimated. To meet Capability Category II, it is required that plant-specific information be collected and considered on a variety of aspects (including material condition of fluid systems, experience with water hammer, and maintenance-induced floods). The current analysis is limited to the use of generic failure rates. This is consistent with Capability Category I. Address potential issues with material condition, experience with water hammer, etc. In particular, further attention should be paid to the possibility of maintenance-induced and other human-caused flooding.	Address potential issues with material condition and water hammer using plant-specific information. Use this information to revise, if necessary, piping failure frequencies available in industry-wide sources, consistent with the Standard. For maintenance-induced and other human-caused flooding, see IFSO-A4. URE 1153 was written to consider updating flood frequency for aging affects based on EPRI-302000079 Rupture frequencies.	Plant specific experience with internal flooding, water hammer is addressed in the IF Notebook rev 1 in Sections 3.3. A discussion of Human-induced floods is contained in Section 5.3. Regarding any effect on flood frequency due to aging affects, a sensitivity evaluation for a particular STI evaluation would show if there was any impact.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IFEV-A7 [2005: IF-D6]	INCLUDE consideration of human-induced floods during maintenance through application of generic data.	complete	**F&O IF-D6-01:** Initiating events that could result from human actions were considered only for a small number of possible maintenance activities. These flood contributors were not evaluated using generic data as required. Operating experience for nuclear power plants has provided evidence that human-caused floods can be important. The SR requires that such floods be evaluated using at least generic data to meet Capability Category I or II. Perform a more detailed assessment of potential human-caused floods, and apply at least generic data to characterize their frequencies.	See IFSO-A4.	Discussion of human caused floods is discussed in detail in Section 3.3 and 5.3 of Internal Flood Notebook (G1-IF-0000-r1) for various systems. Based on the analyses performed, one maintenance induced flood was added to the model, FL-ABO-M-SW – 2,000 gpm SW flood in the Aux Building due to maintenance, isolated within 65 minutes.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IFEV-A8 [2005: IF-D7]	SCREEN OUT flood scenario groups if (a) the quantitative screening criteria in IFSN-A10 (2005 text: IE-C4), as applied to the flood scenario groups, are met; OR (b) the internal flood-initiating event affects only components in a single system, AND it can be shown that the product of the frequency of the flood and the probability of SSC failure given the flood is two orders of magnitude lower than: the product of the non-flooding frequency for the corresponding initiating events in the PRA, AND the random (non-flood-induced) failure probability of the same SSCs that are assumed failed by the flood. If the flood impacts multiple systems, DO NOT screen on this basis.	complete	**F&O IF-D7-01:** Quantitative screening of some scenarios was performed, but it is not clear what criteria were applied in doing so. The criteria should be defined and applied in a clear and consistent manner. SRs IF-D7 and IF-E3a provide explicit criteria for performing quantitative screening of flood scenarios. The IF Notebook documents that some scenarios were screened on low frequency, but does not invoke any particular criteria in doing so. Provide a clear set of criteria for performing quantitative screening of flood scenarios, and apply the criteria in a clear and consistent manner.	Update the Internal Flooding Study (51 - 9100978 - 000) to describe the criteria used to screen flood scenarios. If current screening criteria are not well defined, develop such criteria and apply them to scenarios addressed in the analysis.	No impact to TSTF 425. This issue has been addressed. Internal Flood Notebook Section 4.6, Screening Scenarios and Sources, was updated to document the screening criteria used. Figure 4.1, was added which shows the Screening Criteria and Table 4.6 was edited to show the screening criterion used for various flood scenarios.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IFQU-A5 [2005: IF-E5]	If additional human failure events are required to support quantification of flood scenarios, PERFORM any human reliability analysis in accordance with the applicable requirements described in 2-2.5 (2005 text: Tables 4.5.5-2(e) through 4.5.5-2(h)).	complete	**F&O IF-E5-01:** It was not clear that the requirements were met in all cases. For example, interviews to establish aspects such as response times were apparently performed as part of the flood analysis, but the HRA was dramatically changed and new interviews/changes were not incorporated, nor were any inputs obtained from the HRA performed as part of the flood analysis carried forward. It is necessary to perform the assessment of HFEs associated with internal flooding in the same manner as for other HFEs. The requirements to confirm procedure paths, timing, etc. via interviews with operators were not met for a number of events. Re-examine the HFEs associated with internal flooding, and either perform needed operator interviews or identify and document existing inputs.	Re-examine each HFE included in the flooding analysis. Perform operator interviews as needed or identify and document previously performed interviews. Required operator interviews should comprise the following: 1. evaluate the flooding events based on similarities to identify a select set of scenarios to review with the operators (for example, categorized by the system that generated the flood, e.g., fire protection) 2. schedule interview sessions of about 1/2 hour to an hour per each flooding scenario, conducted separately with two different operators (preferably one experienced, one novice) to get diverse opinions. 3. include questions on timing consistent with the HRA Calculator Time Window screen for time of cue, time to diagnosis, time for execution/manipulation of action (including travel time, with potential flood-related access delays). Be sure to ask about any differences for floods initiated in **same** system but in different rooms. 4. document interviews during the sessions (notes and/or tape recordings) and later in the HRA Calculator screens for Operator Interviews and Time Window. Estimate and document internal flooding HFEs using the same approach as was used for other HFEs in the PRA. Recalculate flood scenario frequencies based on the new HFEs.	No impact to TSTF-425. Ginna Station Flooding Human Reliability Analysis (HRA) documents the flood recovery actions (Areva Document No.: 51-9099406-000 located in GSN 0157). The information and HRA values in this notebook were verified to be consistent with the HRA actions being used in the internal flood model. No additional interviews were identified as being necessary.

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IFQU-B1 [2005: IF-F1]	DOCUMENT the internal flood accident sequences and quantification in a manner that facilitates PRA applications, upgrades, and peer review.	Open	**F&O IF-F1-01:** The documentation is comprised primarily of the internal flooding notebook, supplemented heavily with information provided in a set of Excel worksheets. The notebook is annotated to provide a link to elements of the worksheets, and an "assumption" provides the formal tie between the notebook and the worksheets. Some areas in which the links were indirect or missing were noted. In general, the manner in which important parts of the flood analysis are documented in what would usually be characterized as an informal set of worksheets is judged not to meet the requirement that the analysis be documented in a manner that facilitates applications, upgrades, and peer review. In addition to developing a single integrated set of documentation for the internal flood analysis, there were several areas in which additional documentation would make the analysis more tractable have been provided in connection to other SRs. These include the following: • Include a set of simplified arrangement drawings to explicate the definition of flood areas and help in understanding aspects such as flood propagation. • Tabulate the flood areas and identify clearly which are screened and which retained for further analysis to make the process more tractable. Specify clearly which criteria (qualitative or quantitative) are employed in screening each flood area. • Define explicitly the criteria used to perform quantitative screening as noted in Section 6.0. • Define the criteria used to determine whether a PRA component was susceptible to failure due to spray.	*Documentation only:* Revise the Internal Flooding Study (51 - 9100978 - 000) to meet the documentation requirements of the 2009 Standard. Address NRC Resolutions as appropriate. It is recommended that the Study be reformatted to be consistent with the HLRs and SRs of the Standard, integrating appropriate parts of the worksheets into the primary document. This will provide a document that can be easily reviewed against the standard and easily followed by personnel not involved in the original analysis. Consistent with the F&O, include the following in the revised Study: • Include a set of simplified arrangement drawings to explicate the definition of flood areas and help in understanding aspects such as flood propagation. • Tabulate the flood areas and identify clearly which are screened and which retained for further analysis to make the process more tractable. Specify clearly which criteria (qualitative or quantitative) are employed in screening each flood area. • Define explicitly the criteria used to perform quantitative screening as noted in Section 6.0. • Define the criteria used to determine whether a PRA component was susceptible to failure due to spray.	*This documentation item will not* impact the TSTF 425 analysis. This item has largely been addressed by adding tables in Section 5.2 that show the development of each initiating event frequency, adding an Initiating Event Summary Table (section 5.2.17), adding a simplified set of arrangement drawings showing each flood area (Appendix K), defining spray modeling criteria (Section 3.3.2) and identifying for each flood area whether it was screened and the screening criterion used (Table 4.6). The remaining item is to develop the criteria used to perform quantitative screening, if applicable, in Section 6.0 (URE 1177).

Table 2-1 Internal Events PRA Peer Review – Findings

SR	Topic	Status	Finding/Observation	Disposition	Impact to TSTF-425
IFQU-B3 [2005: IF-F3]	DOCUMENT sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the internal flood accident sequences and quantification. (2005 text: Document the key assumptions and the key sources of uncertainty associated with the internal flooding analysis.)	Open	**F&O IF-F3-01:** Section 7 of the IF Notebook provides a discussion of three areas considered to be major sources of uncertainty in the flood analysis. This does not constitute an adequate characterization of the sources of uncertainty associated with the flood analysis or a comprehensive discussion of the assumptions that could have an effect on the results. A reasonably thorough investigation of sources of uncertainty is necessary for proper characterization of the flood analyses and results. A more comprehensive characterization of sources of uncertainty, comparable to that provided for other areas of the PRA, should be developed for the internal flood analysis.	Documentation only: Update the discussion of assumptions and uncertainty to be consistent with the 2009 Standard. The 2005 Standard required the documentation of key assumptions and key sources of uncertainty, while the 2009 Standard eliminates the term "key." The equivalent sections of other PRA technical elements provide an example of the detail that is required. In addition, the discussion of uncertainty and impact of assumptions in the Quantification Notebook should be revised to include treatment of flood issues (or alternately, a similar treatment should be provided in the Flood Notebook).	This documentation issue will not affect the TSTF 425 analysis. This issue has been partially addressed by the calculation of error factors for the flood initiating events. These have been added to table 5-2, flood frequencies. Remaining action is to reference any key sources of uncertainty from the EPRI guideline on the treatment of uncertainty for ASME PRA Standard SRs related to internal flooding in Section 7 of the PRA Internal Flooding Analysis System Notebook (URE 1178)

ATTACHMENT 3

License Amendment Request

R. E. Ginna Nuclear Power Plant
Docket No. 50-244

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 Page Changes

1.1-4	3.3.6-2	3.5.1-2	3.7.12-1
3.1.1-1	3.4.1-1	3.5.2-2	3.7.14-1
3.1.2-2	3.4.1-2	3.5.2-3	3.8.1-3
3.1.4-3	3.4.2-1	3.5.4-1	3.8.1-4
3.1.5-1	3.4.3-2	3.6.2-4	3.8.1-5
3.1.6-2	3.4.4-1	3.6.3-6	3.8.3-2
3.1.8-2	3.4.5-2	3.6.3-7	3.8.4-2
3.2.1-3	3.4.6-2	3.6.4-1	3.8.6-1
3.2.1-4	3.4.7-2	3.6.5-1	3.8.6-2
3.2.2-2	3.4.8-2	3.6.6-2	3.8.7-2
3.2.3-1	3.4.9-1	3.6.6-3	3.8.8-2
3.2.4-3	3.4.11-3	3.7.2-2	3.8.9-2
3.3.1-8	3.4.12-4	3.7.4-1	3.8.10-2
3.3.1-9	3.4.12-5	3.7.5-3	3.9.1-1
3.3.1-10	3.4.13-2	3.7.6-1	3.9.2-2
3.3.2-3	3.4.14-2	3.7.7-2	3.9.3-2
3.3.2-4	3.4.14-3	3.7.8-2	3.9.4-2
3.3.3-2	3.4.15-3	3.7.9-2	3.9.5-2
3.3.4-2	3.4.16-2	3.7.10-1	3.9.6-1
3.3.5-3	3.5.1-1	3.7.11-1	5.5-13

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission (NRC).
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the plant specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve lift settings and enable temperature associated with the Low Temperature Overpressurization Protection System for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these limits is addressed in individual specifications.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the highest average nuclear power in any quadrant to the average nuclear power in the four quadrants.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1775 MWt.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCAs not capable of being fully inserted, the reactivity worth of the RCCAs must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal hot zero power temperature.
~~STAGGERED TEST BASIS~~	~~A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.~~

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits specified in the COLR.

APPLICABILITY: MODE 2 with k_{eff} < 1.0,
MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM is within the limits specified in the COLR.	~~24 hours~~ INSERT 1

	SURVEILLANCE	FREQUENCY
SR 3.1.2.2	- NOTE - 1. Only required after 60 effective full power days (EFPD). 2. The predicted reactivity values must be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. Verify measured core reactivity is within ± 1% Δk/k of predicted values.	INSERT 1 ~~31 EFPD~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	~~12 hours~~ INSERT 1
SR 3.1.4.2	- NOTE - Only required to be performed if the rod position deviation monitor is inoperable. Verify individual rod positions within alignment limit.	Once within 4 hours and ~~every 4 hours thereafter~~ INSERT 1
SR 3.1.4.3	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core to a MRPI transition in either direction.	~~92 days~~ INSERT 1
SR 3.1.4.4	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 1.8 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500°F$; and b. Both reactor coolant pumps operating.	Once prior to reactor criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limit

LCO 3.1.5 The shutdown bank shall be at or above the insertion limit specified in the COLR.

- NOTE -

The shutdown bank may be outside the limit when required for performance of SR 3.1.4.3.

APPLICABILITY: MODE 1,
MODE 2 with $K_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Shutdown bank not within limit.	A.1.1 Verify SDM is within the limits specified in the COLR. OR A.1.2 Initiate boration to restore SDM to within limit.	1 hour 1 hour
	AND A.2 Restore shutdown bank to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2 with K_{eff} < 1.0.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify the shutdown bank insertion is within the limit specified in the COLR.	~~12 hours~~ INSERT 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.6.1	Verify estimated critical control bank position is within the limits specified in the COLR.	Within 4 hours prior to achieving criticality
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	~~12 hours~~ INSERT 1
SR 3.1.6.3	- NOTE - Only required to be performed if the rod insertion limit monitor is inoperable. Verify each control bank insertion is within the limits specified in the COLR.	Once within 4 hours and ~~every 4 hours thereafter~~ INSERT 1
SR 3.1.6.4	Verify each control bank not fully withdrawn from the core is within the sequence and overlap limits specified in the COLR.	~~12 hours~~ INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Perform a COT on power range and intermediate range channels per SR 3.3.1.7 and SR 3.3.1.8.	Once within 7 days prior to criticality
SR 3.1.8.2	Verify the RCS lowest loop average temperature is ≥ 530°F.	~~30 minutes~~ INSERT 1
SR 3.1.8.3	Verify THERMAL POWER is ≤ 5% RTP.	~~30 minutes~~ INSERT 1
SR 3.1.8.4	Verify SDM is within the limits specified in the COLR.	~~24 hours~~ INSERT 1

SURVEILLANCE REQUIREMENTS

- NOTE -

During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q^C(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified <u>AND</u> ~~31 EFPD thereafter~~

INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	- NOTE - If measurements indicate that the maximum over z $[F_Q^C(Z) / K(Z)]$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits or b. Repeat SR 3.2.1.2 once per 7 EFPD until either a. above is met or two successive flux maps indicate that the maximum over z $[F_Q^C(Z) / K(Z)]$ has not increased.	
	Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified <u>AND</u> ~~31 EFPD thereafter~~

INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND ~~31 EFPD thereafter~~ INSERT 1
SR 3.2.2.2	- NOTE - *Only required to be performed if one power range channel is inoperable with THERMAL POWER ≥ 75% RTP.* Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once within 24 hours and ~~every 24 hours thereafter~~ INSERT 1

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits spcified in the COLR.

- NOTE -

The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	~~7 days~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	- NOTE - 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. Verify QPTR is within limit by calculation.	INSERT 1 ~~7 days~~
SR 3.2.4.2	- NOTE - Not required to be performed until 24 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.	INSERT 1 ~~24 hours~~

CONDITION	REQUIRED ACTION		COMPLETION TIME
	W.2	Restore trip mechanism or train to OPERABLE status.	48 hours
X. Required Action and associated Completion Time of Condition W not met.	X.1	Initiate action to fully insert all rods.	Immediately
	<u>AND</u>		
	X.2	Place the Control Rod Drive System in a Condition incapable of rod withdrawal.	1 hour

SURVEILLANCE REQUIREMENTS

- NOTE -

Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	~~12 hours~~ INSERT 1
SR 3.3.1.2	- NOTE - Required to be performed within 12 hours after THERMAL POWER is ≥ 50% RTP. Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output and adjust if calorimetric power is > 2% higher than indicated NIS power.	~~24 hours~~ INSERT 1
SR 3.3.1.3	- NOTE - 1. Required to be performed within 7 days after THERMAL POWER is ≥ 50% RTP but prior to exceeding 90% RTP following each refueling and if the Surveillance has not been performed within the last 31 EFPD. 2. Performance of SR 3.3.1.6 satisfies this SR. Compare results of the incore detector measurements to NIS AFD and adjust if absolute difference is ≥ 3%.	INSERT 1 ~~31 effective full power days (EFPD)~~

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	Perform TADOT.	~~31 days on a STAGGERED TEST BASIS~~ INSERT 1
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	~~31 days on a STAGGERED TEST BASIS~~ INSERT 1
SR 3.3.1.6	- NOTE - Not required to be performed until 7 days after THERMAL POWER is ≥ 50% RTP, but prior to exceeding 90% RTP following each refueling. Calibrate excore channels to agree with incore detector measurements.	INSERT 1 ~~92 EFPD~~
SR 3.3.1.7	- NOTE - Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entering MODE 3. Perform COT.	INSERT 1 ~~92 days~~
SR 3.3.1.8	- NOTE - 1. Not required for power range and intermediate range instrumentation until 4 hours after reducing power < 6% RTP. 2. Not required for source range instrumentation until 4 hours after reducing power < 5E-11 amps. Perform COT.	INSERT 1 ~~92 days~~
SR 3.3.1.9	- NOTE - Setpoint verification is not required. Perform TADOT.	INSERT 1 ~~92 days~~

	SURVEILLANCE	FREQUENCY
SR 3.3.1.10	- NOTE - Neutron detectors are excluded. Perform CHANNEL CALIBRATION.	INSERT 1 ~~24 months~~
SR 3.3.1.11	Perform TADOT.	~~24 months~~ INSERT 1
SR 3.3.1.12	- NOTE - Setpoint verification is not required. Perform TADOT.	Prior to reactor startup if not performed within previous 31 days
SR 3.3.1.13	Perform COT.	~~24 months~~ INSERT 1

CONDITION	REQUIRED ACTION		COMPLETION TIME
L. As required by Required Action A.1 and referenced by Table 3.3.2-1.	L.1	- NOTE - The inoperable channel may be bypassed for up to 4 hours for surveillance testing of the other channels. Place channel in trip.	6 hours
M. Required Action and associated Completion Time of Condition L not met.	M.1	Be in MODE 3.	6 hours
	AND		
	M.2	Reduce pressurizer pressure to < 2000 psig.	12 hours
N. As required by Required Action A.1 and referenced by Table 3.3.2-1.	N.1	Declare associated Auxiliary Feedwater pump inoperable and enter applicable condition(s) of LCO 3.7.5, "Auxiliary Feedwater (AFW) System."	Immediately

SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	~~12 hours~~ INSERT 1
SR 3.3.2.2	Perform COT.	~~92 days~~ INSERT 1
SR 3.3.2.3	- NOTE - Verification of relay setpoints not required. Perform TADOT.	~~92 days~~ INSERT 1

	SURVEILLANCE	FREQUENCY
SR 3.3.2.4	- NOTE - Verification of relay setpoints not required. Perform TADOT.	INSERT 1 ~~24 months~~
SR 3.3.2.5	Perform CHANNEL CALIBRATION.	~~24 months~~ INSERT 1
SR 3.3.2.6	Verify the Pressurizer Pressure-Low and Steam Line Pressure-Low Functions are not bypassed when pressurizer pressure > 2000 psig.	~~24 months~~ INSERT 1
SR 3.3.2.7	Perform ACTUATION LOGIC TEST.	~~24 months~~ INSERT 1

CONDITION		REQUIRED ACTION		COMPLETION TIME
D.	One or more Functions with two required channels inoperable.	D.1	Restore one channel to OPERABLE status.	7 days
E.	Required Action and associated Completion Time of Condition C or D not met.	E.1	Enter the Condition referenced in Table 3.3.3-1 for the channel.	Immediately
F.	As required by Required Action E.1 and referenced in Table 3.3.3-1.	F.1	Be in MODE 3.	6 hours
		AND		
		F.2	Be in MODE 4.	12 hours
G.	As required by Required Action E.1 and referenced in Table 3.3.3-1.	G.1	Initiate action to prepare and submit a special report.	Immediately

SURVEILLANCE REQUIREMENTS

- NOTE -
SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	~~31 days~~ INSERT 1
SR 3.3.3.2	Perform CHANNEL CALIBRATION.	~~24 months~~ INSERT 1

SURVEILLANCE REQUIREMENTS

- NOTE -

When a channel is placed in an inoperable status solely for the performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 4 hours provided the second channel maintains LOP DG start capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform TADOT.	~~31 days~~ INSERT 1
SR 3.3.4.2	Perform CHANNEL CALIBRATION with Limiting Safety System Settings (LSSS)[a] for each 480 V bus as follows: a. Loss of voltage LSSS ≥ 372.0 V and ≤ 374.8 V with a time delay of ≥ 2.13 seconds and ≤ 2.62 seconds. b. Degraded voltage LSSS ≥ 420.0 V and ≤ 423.6 V with a time delay of ≥ 68.1 seconds and ≤ 125 seconds (@ 420 V) and ≥ 71.8 seconds and ≤ 125 seconds (@ 423.6 V).	~~24 months~~ INSERT 1

(a)

A channel is OPERABLE when both of the following conditions are met:

1. The absolute difference between the as-found Trip Setpoint (TSP) and the previous as-left TSP is within the CHANNEL CALIBRATION Acceptance Criteria. The CHANNEL CALIBRATION Acceptance Criteria is defined as:

   |as-found TSP - previous as-left TSP| ≤ CHANNEL CALIBRATION uncertainty

   The CHANNEL CALIBRATION uncertainty shall not include the calibration tolerance.

2. The as-left TSP is within the established calibration tolerance band about the nominal TSP. The nominal TSP is the desired setting and shall not exceed the LSSS. The LSSS and the established calibration tolerance band are defined in accordance with the Ginna Instrument Setpoint Methodology. The channel is considered operable even if the as-left TSP is non-conservative with respect to the LSSS provided that the as-left TSP is within the established calibration tolerance band.

SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.5-1 to determine which SRs apply for each Containment Ventilation Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK.	~~24 hours~~ INSERT 1
SR 3.3.5.2	Perform COT.	~~92 days~~ INSERT 1
SR 3.3.5.3	Perform ACTUATION LOGIC TEST.	~~24 months~~ INSERT 1
SR 3.3.5.4	Perform CHANNEL CALIBRATION.	~~24 months~~ INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.6-1 to determine which SRs apply for each CREATS Actuation Function.

SURVEILLANCE		FREQUENCY INSERT 1
SR 3.3.6.1	Perform CHANNEL CHECK.	~~12 hours~~ INSERT 1
SR 3.3.6.2	Perform COT.	~~92 days~~ INSERT 1
SR 3.3.6.3	- NOTE - Verification of setpoint is not required. Perform TADOT.	INSERT 1 ~~24 months~~
SR 3.3.6.4	Perform CHANNEL CALIBRATION.	INSERT 1 ~~24 months~~
SR 3.3.6.5	Perform ACTUATION LOGIC TEST.	~~24 months~~ INSERT 1

3.4 REACTOR COOLANT SYSTEMS (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR.

- NOTE -

Pressurizer pressure limit does not apply during pressure transients due to:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is within limit specified in the COLR.	~~12 hours~~ INSERT 1
SR 3.4.1.2	Verify RCS average temperature is within limit specified in the COLR.	~~12 hours~~ INSERT 1

	SURVEILLANCE	FREQUENCY
SR 3.4.1.3	- NOTE - Required to be performed within 7 days after ≥ 95% RTP. Verify RCS total flow rate is within the limit specified in the COLR.	INSERT 1 ~~24 months~~

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ 540°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or both RCS loops not within limit.	A.1 Be in MODE 2 with K_{eff} < 1.0.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS T_{avg} in each loop ≥ 540°F.	Within 30 minutes prior to achieving criticality.
SR 3.4.2.2	- NOTE - Only required if any RCS loop T_{avg} < 547°F and the low T_{avg} alarm is either inoperable or not reset. Verify RCS T_{avg} in each loop ≥ 540°F.	Once within 30 minutes and ~~every 30 minutes thereafter~~ INSERT 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.3.1	- NOTE - Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	INSERT 1 ~~30 minutes~~

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODE 1 > 8.5% RTP

LCO 3.4.4 Two RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODE 1 > 8.5% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 1 ≤ 8.5% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify each RCS loop is in operation.	~~12 hours~~ INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Both RCS loops inoperable. OR No RCS loop in operation.	C.1 De-energize all CRDMs. AND C.2 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1. AND C.3 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loop is in operation.	~~12 hours~~
SR 3.4.5.2	Verify steam generator secondary side water levels are ≥ 16% for two RCS loops.	~~12 hours~~ INSERT 1
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required RCP that is not in operation.	~~7 days~~ INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One RHR loop inoperable. AND Two RCS loops inoperable.	- NOTE - Required Action B.1 is not applicable if all RCS and RHR loops are inoperable and Condition C is entered. B.1 Be in MODE 5.	 24 hours
C. All RCS and RHR loops inoperable. OR No RCS or RHR loop in operation.	C.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1. AND C.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	~~12 hours~~
SR 3.4.6.2	Verify SG secondary side water level is ≥ 16% for each required RCS loop.	~~12 hours~~ INSERT 1
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	~~7 days~~ INSERT 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable. AND Both SGs secondary side water levels not within limits.	A.1 Initiate action to restore a second RHR loop to OPERABLE status. OR A.2 Initiate action to restore required SG secondary side water levels to within limits.	Immediately Immediately
B. Both RHR loops inoperable. OR No RHR loop in operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1. AND B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one RHR loop is in operation.	~~12 hours~~
SR 3.4.7.2	Verify SG secondary side water level is ≥ 16% in the required SG.	~~12 hours~~ INSERT 1
SR 3.4.7.3	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	~~7 days~~ INSERT

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.2	Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify one RHR loop is in operation.	~~12 hours~~
SR 3.4.8.2	Verify correct breaker alignment and indicated power are available to the RHR pump that is not in operation.	~~7 days~~ INSERT 1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open. AND A.2 Be in MODE 4.	6 hours 12 hours
B. Pressurizer heaters capacity not within limits.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ 87%.	~~12 hours~~ INSERT 1
SR 3.4.9.2	Verify total capacity of the pressurizer heaters is ≥ 100 Kw.	~~92 days~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	- NOTE - Not required to be performed with block valve closed per LCO 3.4.13. Perform a complete cycle of each block valve.	INSERT 1 ~~92 days~~
SR 3.4.11.2	Perform a complete cycle of each PORV.	~~24 months~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	- NOTE - Only required to be performed when complying with LCO 3.4.12.a. Verify no SI pump is capable of injecting into the RCS.	INSERT 1 ~~12 hours~~
SR 3.4.12.2	- NOTE - Only required to be performed when complying with LCO 3.4.12.b. Verify a maximum of one SI pump is capable of injecting into the RCS.	INSERT 1 ~~12 hours~~
SR 3.4.12.3	- NOTE - Only required to be performed when ECCS accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. Verify each ECCS accumulator motor operated isolation valve is closed.	Once within 12 hours and ~~every 12 hours thereafter~~ INSERT 1
SR 3.4.12.4	- NOTE - Only required to be performed when complying with LCO 3.4.12.b. Verify RCS vent ≥ 1.1 square inches open.	INSERT 1 ~~12 hours~~ for unlocked open vent valve(s) <u>AND</u> INSERT 1 ~~31 days~~ for locked open vent valve(s)
SR 3.4.12.5	Verify PORV block valve is open for each required PORV.	~~72 hours~~ INSERT 1

	SURVEILLANCE	FREQUENCY
SR 3.4.12.6	- NOTE - Required to be performed within 12 hours after decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature specified in the PTLR. Perform a COT on each required PORV, excluding actuation.	INSERT 1 ~~31 days~~
SR 3.4.12.7	- NOTE - Only required to be performed when ECCS accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. Verify power is removed from each ECCS accumulator motor operated isolation valve operator.	Once within 12 hours and ~~every 31 days thereafter~~ INSERT 1
SR 3.4.12.8	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	~~24 months~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	- NOTE - 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	INSERT 1 ~~72 hours~~
SR 3.4.13.2	- NOTE - Not required to be performed until 12 hours after establishment of steady state operation. Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	INSERT 1 ~~72 hours~~

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	- NOTE - 1. Not required to be performed until prior to entering MODE 2 from MODE 3. 2. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. Verify leakage from each SI cold leg injection line and each RHR RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.	INSERT 1 ~~24 months~~ AND Within 24 hours following valve actuation due to automatic or manual action, flow through the valve, or maintenance on the valve

SURVEILLANCE		FREQUENCY
SR 3.4.14.2	- NOTE - 1. Not required to be performed until prior to entering MODE 2 from MODE 3. 2. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. Verify leakage from each SI hot leg injection line RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.	INSERT 1 ~~40 months~~ AND Within 24 hours following valve actuation due to automatic or manual action, flow through the valve, or maintenance on the valve

SURVEILLANCE REQUIREMENTS

INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of containment atmosphere radioactivity monitors.	~~12 hours~~ INSERT 1
SR 3.4.15.2	Perform COT of containment atmosphere radioactivity monitors.	~~92 days~~ INSERT 1
SR 3.4.15.3	Perform CHANNEL CALIBRATION of the required containment sump monitor.	~~24 months~~ INSERT 1
SR 3.4.15.4	Perform CHANNEL CALIBRATION of containment atmosphere radioactivity monitors.	~~24 months~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/$\bar{E}$ μCi/gm.	~~7 days~~ INSERT 1
SR 3.4.16.2	- NOTE - Only required to be performed in MODE 1. Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	INSERT 1 ~~14 days~~ AND Between 2 and 10 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period
SR 3.4.16.3	- NOTE - Only required to be performed in MODE 1. Determine $\bar{E}$ from a reactor coolant sample.	Once within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. AND INSERT 1 ~~Every 184 days thereafter~~

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Two ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure > 1600 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce pressurizer pressure to ≤ 1600 psig.	6 hours 12 hours
D. Two accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator motor operated isolation valve is fully open.	~~12 hours~~ INSERT 1
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 1090 cubic feet (24%) and ≤ 1140 cubic feet (83%).	~~12 hours~~ INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 700 psig and ≤ 790 psig.	INSERT 1 ~~12 hours~~
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2550 ppm and ≤ 3050 ppm.	INSERT 1 ~~12 hours~~ (by inleakage monitoring) AND INSERT 1 ~~6 months~~ (by sample)
SR 3.5.1.5	Verify power is removed from each accumulator motor operated isolation valve operator when pressurizer pressure is > 1600 psig.	INSERT 1 ~~31 days~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position. Number / Position / Function 825A Open RWST Suction to SI Pumps 825B Open RWST Suction to SI Pumps 826A Closed BAST Suction to SI Pumps 826B Closed BAST Suction to SI Pumps 826C Closed BAST Suction to SI Pumps 826D Closed BAST Suction to SI Pumps 851A Open Sump B to RHR Pumps 851B Open Sump B to RHR Pumps 856 Open RWST Suction to RHR Pumps 878A Closed SI Injection to RCS Hot Leg 878B Open SI Injection to RCS Cold Leg 878C Closed SI Injection to RCS Hot Leg 878D Open SI Injection to RCS Cold Leg 896A Open RWST Suction to SI and Containment Spray 896B Open RWST Suction to SI and Containment Spray	~~12 hours~~ INSERT 1
SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	INSERT 1 ~~31 days~~
SR 3.5.2.3	Verify each breaker or key switch, as applicable, for each valve listed in SR 3.5.2.1, is in the correct position.	INSERT 1 ~~31 days~~

	SURVEILLANCE	FREQUENCY
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	~~24 months~~ INSERT 1
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	INSERT 1 ~~24 months~~
SR 3.5.2.7	Verify, by visual inspection, each RHR containment sump suction inlet is not restricted by debris and the containment sump screen shows no evidence of structural distress or abnormal corrosion.	~~24 months~~ INSERT 1

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. RWST water volume not within limits.	B.1 Restore RWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify RWST borated water volume is ≥ 300,000 gallons (88%).	~~7 days~~ INSERT 1
SR 3.5.4.2	Verify RWST boron concentration is ≥ 2750 ppm and ≤ 3050 ppm.	~~7 days~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	- NOTE - 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in each air lock can be opened at a time.	~~24 months~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each mini-purge valve is closed, except when the penetration flowpath(s) are permitted to be open under administrative control.	~~31 days~~ INSERT 1
SR 3.6.3.2	- NOTE - 1. Isolation boundaries in high radiation areas may be verified by use of administrative controls. 2. Not applicable to containment isolation boundaries which receive an automatic containment isolation signal. Verify each containment isolation boundary that is located outside containment and not locked, sealed, or otherwise secured in the required position is performing its containment isolation accident function except for containment isolation boundaries that are open under administrative controls.	INSERT 1 ~~92 days~~
SR 3.6.3.3	- NOTE - 1. Isolation boundaries in high radiation areas may be verified by use of administrative means. 2. Not applicable to containment isolation boundaries which receive an automatic containment isolation signal. Verify each containment isolation boundary that is located inside containment and not locked, sealed, or otherwise secured in the required position is performing its containment isolation accident function, except for containment isolation boundaries that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.5	Perform required leakage rate testing of containment mini-purge valves with resilient seals in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Program.

SURVEILLANCE		FREQUENCY
SR 3.6.3.6	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in the required position actuates to the isolation position on an actual or simulated actuation signal.	~~24 months~~ INSERT 1

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -2.0 psig and ≤ 1.0 psig.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	~~12 hours~~ INSERT 1

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 125°F.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	~~12 hours~~ INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two CS trains inoperable. <u>OR</u> Three or more CRFC units inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Perform SR 3.5.2.1 and SR 3.5.2.3 for valves 896A and 896B.	In accordance with applicable SRs.
SR 3.6.6.2	Verify each CS manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	~~31 days~~ INSERT 1
SR 3.6.6.3	Verify each NaOH System manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	~~31 days~~ INSERT 1
SR 3.6.6.4	Operate each CRFC unit for ≥ 15 minutes.	~~31 days~~ INSERT 1
SR 3.6.6.5	Verify cooling water flow through each CRFC unit.	~~31 days~~ INSERT 1
SR 3.6.6.6	Verify each CS pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.7	Verify NaOH System solution volume is ≥ 3000 gal.	~~184 days~~ INSERT 1
SR 3.6.6.8	Verify NaOH System tank NaOH solution concentration is ≥ 30% and ≤ 35% by weight.	~~184 days~~ INSERT 1
SR 3.6.6.9	Perform required CRFC unit testing in accordance with the VFTP.	In accordance with the VFTP
SR 3.6.6.10	Verify each automatic CS valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	~~24 months~~ INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.6.6.11	Verify each CS pump starts automatically on an actual or simulated actuation signal.	~~24 months~~ INSERT 1
SR 3.6.6.12	Verify each CRFC unit starts automatically on an actual or simulated actuation signal.	~~24 months~~ INSERT 1
SR 3.6.6.13	Verify each automatic NaOH System valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	~~24 months~~ INSERT 1
SR 3.6.6.14	Verify spray additive flow through each eductor path.	INSERT 1 ~~5 years~~
SR 3.6.6.15	Verify each spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify closure time of each MSIV is ≤ 5 seconds under no flow and no load conditions.	In accordance with the Inservice Testing Program
SR 3.7.2.2	Verify each main steam non-return check valve can close.	In accordance with the Inservice Testing Program
SR 3.7.2.3	Verify each MSIV can close on an actual or simulated actuation signal.	~~24 months~~ $\boxed{\text{INSERT 1}}$

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Relief Valves (ARVs)

LCO 3.7.4 Two ARV lines shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System average temperature (T_{avg}) $\geq$ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ARV line inoperable.	A.1 Restore ARV line to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3 with T_{avg} < 500°F.	8 hours
C. Two ARV lines inoperable.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS. INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Perform a complete cycle of each ARV.	~~24 months~~
SR 3.7.4.2	Verify one complete cycle of each ARV block valve.	~~24 months~~

INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each AFW and SAFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~ INSERT 1
SR 3.7.5.2	- NOTE - Required to be met prior to entering MODE 1 for the TDAFW pump.	
	Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.3	Verify the developed head of each SAFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.7.5.4	Perform a complete cycle of each AFW and SAFW motor operated suction valve from the Service Water System, each AFW and SAFW discharge motor operated isolation valve, and each SAFW cross-tie motor operated valve.	In accordance with the Inservice Testing Program
SR 3.7.5.5	Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~24 months~~ INSERT 1
SR 3.7.5.6	- NOTE - Required to be met prior to entering MODE 1 for the TDAFW pump.	
	Verify each AFW pump starts automatically on an actual or simulated actuation signal.	INSERT 1 ~~24 months~~
SR 3.7.5.7	Verify each SAFW train can be actuated and controlled from the control room.	~~24 months~~ INSERT 1

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tanks (CSTs)

LCO 3.7.6 The CSTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CST water volume not within limit.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours
	AND	
	A.2 Restore CST water volume to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the CST water volume is ≥ 24,350 gal.	~~12 hours~~ INSERT 1

CONDITION	REQUIRED ACTION		COMPLETION TIME
	D.2	Be in MODE 3.	6 hours
	AND		
	D.3	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.7.1	- NOTE - Isolation of CCW flow to individual components does not render the CCW loop header inoperable. Verify each CCW manual and power operated valve in the CCW train and heat exchanger flow path and loop header that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~ INSERT 1
SR 3.7.7.2	Perform a complete cycle of each motor operated isolation valve to the residual heat removal heat exchangers.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify screenhouse bay water level and temperature are within limits.	~~24 hours~~ INSERT 1
SR 3.7.8.2	- NOTE - Isolation of SW flow to individual components does not render the SW loop header inoperable. Verify each SW manual, power operated, and automatic valve in the SW flow path and loop header that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31 days~~ INSERT 1
SR 3.7.8.3	Verify all SW loop header cross-tie valves are locked in the correct position.	~~31 days~~ INSERT 1
SR 3.7.8.4	Verify each SW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~24 months~~ INSERT 1
SR 3.7.8.5	Verify each SW pump starts automatically on an actual or simulated actuation signal.	~~24 months~~ INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	D.1 Place OPERABLE CREATS train in emergency mode. OR D.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
E. Two CREATS trains inoperable during movement of irradiated fuel assemblies. OR One or more CREATS trains inoperable due to an inoperable CRE boundary during movement of irradiated fuel assemblies.	E.1 Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CREATS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate each CREATS filtration train ≥ 15 minutes.	~~31 days~~
SR 3.7.9.2	Perform required CREATS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.9.3	Verify each CREATS train actuates on an actual or simulated actuation signal.	~~24 months~~ INSERT 1

3.7 PLANT SYSTEMS

3.7.10 Auxiliary Building Ventilation System (ABVS)

LCO 3.7.10 The ABVS shall be OPERABLE and in operation.

APPLICABILITY: During movement of irradiated fuel assemblies in the Auxiliary Building when one or more fuel assemblies in the Auxiliary Building has decayed < 60 days since being irradiated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ABVS inoperable.	A.1 - NOTE - LCO 3.0.3 is not applicable. Suspend movement of irradiated fuel assemblies in the Auxiliary Building.	Immediately

SURVEILLANCE REQUIREMENTS

INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify ABVS is in operation.	~~24 hours~~
SR 3.7.10.2	Verify ABVS maintains a negative pressure with respect to the outside environment at the Auxiliary Building operating floor level.	~~24 hours~~ INSERT 1
SR 3.7.10.3	Perform required Spent Fuel Pool Charcoal Adsorber System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

3.7 PLANT SYSTEMS

3.7.11 Spent Fuel Pool (SFP) Water Level

LCO 3.7.11 The SFP water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the SFP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP water level not within limit.	A.1 ------------------ - NOTE - LCO 3.0.3 is not applicable. ------------------ Suspend movement of irradiated fuel assemblies in the SFP.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify the SFP water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	~~7 days~~ INSERT 1

3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool (SFP) Boron Concentration

LCO 3.7.12 The SFP boron concentration shall be ≥ 2300 ppm.

APPLICABILITY: Whenever any fuel assembly is stored in the SFP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP boron concentration not within limit.	- NOTE - LCO 3.0.3 is not applicable.	
	A.1 Suspend movement of fuel assemblies in the SFP.	Immediately
	AND	
	A.2 Initiate action to restore SFP boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify the SFP pool boron concentration is within limit.	~~7 days~~ INSERT 1

3.7 PLANT SYSTEMS

3.7.14 Secondary Specific Activity

LCO 3.7.14 The specific activity of the secondary coolant shall be $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the specific activity of the secondary coolant is $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131.	~~31 days~~ INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours
E. Two DGs inoperable.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for the offsite circuit to each of the 480 V safeguards buses.	~~7 days~~ INSERT 1
SR 3.8.1.2	- NOTE - 1. Performance of SR 3.8.1.9 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. Verify each DG starts from standby conditions and achieves rated voltage and frequency.	INSERT 1 ~~31 days~~

	SURVEILLANCE	FREQUENCY
SR 3.8.1.3	- NOTE - 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.9. Verify each DG is synchronized and loaded and operates for ≥ 60 minutes and < 120 minutes at a load ≥ 2025 kW and < 2250 kW.	INSERT 1 ~~31 days~~
SR 3.8.1.4	Verify the fuel oil level in each day tank.	INSERT 1 ~~31 days~~
SR 3.8.1.5	Verify the DG fuel oil transfer system operates to transfer fuel oil from each storage tank to the associated day tank.	~~31 days~~ INSERT 1
SR 3.8.1.6	Verify transfer of AC power sources from the 50/50 mode to the 100/0 mode and 0/100 mode.	~~24 months~~ INSERT 1
SR 3.8.1.7	- NOTE - 1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. Verify each DG does not trip during and following a load rejection of ≥ 295 kW.	INSERT 1 ~~24 months~~

	SURVEILLANCE	FREQUENCY
SR 3.8.1.8	- NOTE - 1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. Verify each DG automatic trips are bypassed on an actual or simulated safety injection (SI) signal except: a. Engine overspeed; b. Low lube oil pressure; and c. Start failure (overcrank) relay.	~~24 months~~ INSERT 1
SR 3.8.1.9	- NOTE - 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 3. Credit may be taken for unplanned events that satisfy this SR. Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated SI actuation signal: a. De-energization of 480 V safeguards buses; b. Load shedding from 480 V safeguards buses; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads, 2. energizes auto-connected emergency loads through the load sequencer, and 3. supplies permanently and auto-connected emergency loads for $\geq$ 5 minutes.	INSERT 1 ~~24 months~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ 5000 gal of diesel fuel oil for each required DG.	~~31 days~~ INSERT 1
SR 3.8.3.2	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 129 V on float charge.	~~7 days~~ INSERT 1
SR 3.8.4.2	- NOTE - 1. SR 3.8.4.3 may be performed in lieu of SR 3.8.4.2. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	INSERT 1 ~~24 months~~
SR 3.8.4.3	- NOTE - This Surveillance shall not be performed in MODE 1, 2, 3, or 4. Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test.	INSERT 1 ~~60 months~~ AND 12 months when battery shows degradation, or has reached 85% of expected life with capacity < 100% of manufacturer's rating AND 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for Train A and Train B batteries shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DC electrical power sources are required to be OPERABLE by LCO 3.8.5, "DC Sources - MODES 5 and 6."

ACTIONS

- NOTE -
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within limits.	A.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.6.1	Verify electrolyte level of each connected battery cell is above the top of the plates and not overflowing.	~~31 days~~ INSERT 1
SR 3.8.6.2	Verify the float voltage of each connected battery cell is > 2.07 V.	~~31 days~~ INSERT 1
SR 3.8.6.3	Verify specific gravity of the designated pilot cell in each battery is ≥ 1.195.	~~31 days~~ INSERT 1
SR 3.8.6.4	Verify average electrolyte temperature of the designated pilot cell in each battery is ≥ 55°F.	~~31 days~~ INSERT 1
SR 3.8.6.5	Verify average electrolyte temperature of every fifth cell of each battery is ≥ 55°F.	~~92 days~~ INSERT 1

SURVEILLANCE		FREQUENCY
SR 3.8.6.6	Verify specific gravity of each connected battery cell is: a. Not more than 0.020 below average of all connected cells, and b. Average of all connected cells is ≥ 1.195.	~~92 days~~ INSERT 1

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two or more required instrument bus sources inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct static switch alignment to Instrument Bus A and C.	~~7 days~~ INSERT 1
SR 3.8.7.2	Verify correct Class 1E CVT alignment to Instrument Bus B.	~~7 days~~ INSERT 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.8.1	Verify correct static switch alignment to required AC instrument bus(es).	~~7 days~~ INSERT 1
SR 3.8.8.2	Verify correct Class 1E CVT alignment to the required AC instrument bus.	~~7 days~~ INSERT 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required electrical power trains.	~~7 days~~ INSERT 1

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2.1	Suspend CORE ALTERATIONS.	Immediately
		AND	
	A.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
		AND	
	A.2.3	Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
		AND	
	A.2.4	Initiate actions to restore required electrical power distribution train(s) to OPERABLE status.	Immediately
		AND	
	A.2.5	Declare associated required residual heat removal loop(s) inoperable and not in operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	Verify correct breaker alignments and voltage to required electrical power distribution trains.	~~7 days~~ INSERT 1

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	~~72 hours~~ INSERT 1

CONDITION	REQUIRED ACTION		COMPLETION TIME
	C.2	Suspend positive reactivity additions.	Immediately
	AND		
	C.3	Perform SR 3.9.1.1	4 hours AND Once per 12 hours thereafter

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	~~12 hours~~ INSERT 1
SR 3.9.2.2	- NOTE - Neutron detectors are excluded from CHANNEL CALIBRATION. Perform CHANNEL CALIBRATION.	INSERT 1 ~~24 months~~

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	~~7 days~~ INSERT 1
SR 3.9.3.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	~~24 months~~ INSERT 1

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.4	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one RHR loop is in operation and circulating reactor coolant.	~~12 hours~~ INSERT 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation and circulating reactor coolant.	~~12 hours~~ INSERT 1
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	~~7 days~~ INSERT 1

3.9 REFUELING OPERATIONS

3.9.6 Refueling Cavity Water Level

LCO 3.9.6 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment,
During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	~~24 hours~~ INSERT 1

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 SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability and determining CRE unfiltered inleakage as required by paragraph c.

INSERT 2

ATTACHMENT 4

License Amendment Request

R. E. Ginna Nuclear Power Plant
Docket No. 50-244

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)

<u>Proposed Technical Specification Bases Page Changes</u>

(NOTE: TS Bases pages below marked with an asterisk (*) do not contain any mark-ups. These pages are provided for completeness and for information purposes only.)

B 3.1.1-5	B 3.3.1-46	B 3.4.9-4	B 3.6.2-7	B 3.7.14-3	B 3.9.6-3
B 3.1.2-5	B 3.3.1-47	B 3.4.11-7	B 3.6.3-11	B 3.8.1-12	
B 3.1.4-8	B 3.3.2-31	B 3.4.12-10*	B 3.6.3-12	B 3.8.1-13	
B 3.1.4-9	B 3.3.2-32	B 3.4.12-11	B 3.6.3-14	B 3.8.1-14	
B 3.1.5-5	B 3.3.2-33	B 3.4.12-12	B 3.6.4-3	B 3.8.1-15	
B 3.1.6-6	B 3.3.2-34	B 3.4.12-13	B 3.6.5-3	B 3.8.1-16	
B 3.1.8-7	B 3.3.3-16	B 3.4.13-4*	B 3.6.6-8	B 3.8.3-3	
B 3.1.8-8	B 3.3.3-17	B 3.4.13-5	B 3.6.6-9	B 3.8.4-6	
B 3.2.1-9	B 3.3.4-7	B 3.4.13-6	B 3.6.6-10	B 3.8.4-7	
B 3.2.1-10	B 3.3.5-8	B 3.4.14-5*	B 3.6.6-11	B 3.8.4-8	
B 3.2.1-11	B 3.3.5-9	B 3.4.14-6	B 3.7.2-6	B 3.8.6-3	
B 3.2.2-5*	B 3.3.6-7	B 3.4.14-7	B 3.7.4-4	B 3.8.6-4	
B 3.2.2-6	B 3.3.6-8	B 3.4.15-5	B 3.7.5-8	B 3.8.7-6	
B 3.2.3-3	B 3.4.1-4	B 3.4.15-6*	B 3.7.5-9	B 3.8.8-5	
B 3.2.4-5	B 3.4.1-5	B 3.4.16-4	B 3.7.5-10	B 3.8.9-9	
B 3.2.4-6	B 3.4.2-3	B 3.4.16-5	B 3.7.6-3	B 3.8.10-6	
B 3.3.1-39*	B 3.4.3-6	B 3.5.1-6	B 3.7.7-6	B 3.9.1-4	
B 3.3.1-40	B 3.4.4-3	B 3.5.1-7	B 3.7.8-7	B 3.9.2-3*	
B 3.3.1-41	B 3.4.5-5	B 3.5.2-11	B 3.7.8-8	B 3.9.2-4	
B 3.3.1-42	B 3.4.5-6	B 3.5.2-12	B 3.7.9-6	B 3.9.3-4	
B 3.3.1-43	B 3.4.6-5	B 3.5.2-13	B 3.7.10-4	B 3.9.4-4	
B 3.3.1-44	B 3.4.7-5	B 3.5.4-4	B 3.7.11-3	B 3.9.5-4	
B 3.3.1-45	B 3.4.8-4	B 3.5.4-5	B 3.7.12-3	B 3.9.6-2*	

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the flowpath of choice would utilize a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% Δk/k must be recovered and a boration flow rate of 10 gpm using 13,000 ppm boric acid solution, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% Δk/k. These boration parameters of 10 gpm and 13,000 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODE 2 with $K_{eff} < 1.0$ and MODES 3, 4, and 5, theSDM is verified by comparing the RCS boron concentration to a SHUTDOWN MARGIN requirement curve that was generated by taking into account estimated RCS boron concentrations, core power defect, control bank position, RCS average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and isothermal temperature coefficient (ITC).

INSERT 3

~~The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.~~

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27 and 28, Issued for comment July 10, 1967.
2. "American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
3. UFSAR, Section 15.1.5.
4. UFSAR, Section 15.4.4.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, or if the Required Actions of Condition A cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 with $K_{eff} < 1.0$ within 6 hours. If the SDM for MODE 2 with $K_{eff} < 1.0$ is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 with $K_{eff} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity must be verified following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling). The comparison must be made prior to entering MODE 1 when the core conditions such as control rod position, moderator temperature, and samarium concentration are fixed or stable. Since the reactor must be critical to verify core reactivity, it is acceptable to enter MODE 2 with $K_{eff} \geq 1.0$ to perform this SR. This SR is modified by a Note to clarify that the SR does not need to be performed until prior to entering MODE 1.

SR 3.1.2.2

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. ~~The Frequency of 31 EFPD, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.~~ The SR is modified by two Notes. The first Note states that the SR is only required after 60 effective full power days (EFPD). The second Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 EFPD after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.

INSERT 3

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the plant conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the plant must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to at least MODE 2 with $K_{eff} < 1.0$ within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 with $K_{eff} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

~~Verification that individual rod positions are within alignment limits using MRPI or the PPCS at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. This Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.~~ INSERT 3

SR 3.1.4.2

When the rod position deviation monitor (i.e., the PPCS) is inoperable, no control room alarm is available between the normal ~~12 hour~~ Frequency to alert the operators of a rod misalignment. A reduction of the Frequency ~~to 4 hours~~ provides sufficient monitoring of the rod positions when the monitor is inoperable. This Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected. INSERT 3

This SR is modified by a Note that states that performance of this SR is only necessary when the rod position deviation monitor is inoperable.

SR 3.1.4.3

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2 with $K_{eff} \geq 1.0$, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod ~~every 92 days~~ provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod to a MRPI transition will not cause radial or axial power tilts, or oscillations, to occur. INSERT 3 ~~The 92 day Frequency takes into consideration other information~~

~~available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods.~~ During or between required performances of SR 3.1.4.3 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable and aligned, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.4

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with both RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 6, 14, 27, and 28, Issued for comment July 10, 1967.
2. 10 CFR 50.46.
3. UFSAR, Chapter 15.
4. UFSAR, Section 15.4.6.
5. UFSAR, Section 15.1.5.
6. UFSAR, Section 15.4.2.

plant to remain in an unacceptable condition for an extended period of time.

B.1

If Required Actions A.1 and A.2 cannot be completed within the associated Completion Times, the plant must be brought to a MODE where the LCO is not applicable. To achieve this status, the plant must be placed in MODE 2 with k_{eff} < 1.0 within a Completion Time of 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

INSERT 1

Since the shutdown bank is positioned manually by the control room operator, a verification of shutdown bank position at a Frequency ~~of every 12 hours~~ is adequate to ensure that the bank is within the insertion limit. ~~Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.~~

INSERT 3

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 32, Issued for comment July 10, 1967.
2. 10 CFR 50.46.
3. UFSAR, Chapter 15.
4. UFSAR, Section 15.1.5.
5. UFSAR, Section 15.4.1.
6. UFSAR, Section 15.4.2.
7. UFSAR, Section 15.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits. The Frequency of within 4 hours prior to achieving criticality ensures that the estimated control bank position is within the limits specified in the COLR shortly before criticality is reached.

SR 3.1.6.2

~~With an OPERABLE bank insertion limit monitor (i.e., the control board annunciators, verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours.~~ INSERT 3

SR 3.1.6.3

When the insertion limit monitor ~~(i.e., the control board annunciators~~ becomes inoperable, no control room alarm is available ~~between the normal 12 hour frequency~~ to alert the operators of a control bank not within the insertion limits. A reduction of the Frequency to every 4 hours provides sufficient monitoring of control rod insertion when the monitor is inoperable. Verification of the control bank position at a Frequency ~~of 4 hours~~ INSERT 1 is sufficient to detect control banks that may be approaching the insertion limits. INSERT 3

This SR is modified by a Note that states that performance of this SR in only necessary when the rod insertion limit monitor is inoperable.

SR 3.1.6.4

When control banks are maintained within their insertion limits as required by SR 3.1.6.2 and SR 3.1.6.3 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. ~~A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.~~ INSERT 3

D.1

If Required Action C.1 cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 from MODE 2 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel within 7 days prior to criticality. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 7 day time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is ≥ 530°F will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Control board indication for T_{avg} is available down to 540°F while indication from the plant process computer (PPCS) is available down to 535°F. Between 530°F and 535°F, PPCS cold and hot leg indication should be used to determine T_{avg}.

~~Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.~~

INSERT 3

SR 3.1.8.3

Verification that THERMAL POWER is ≤ 5% RTP using the NIS detectors will ensure that the plant is not operating in a condition that could invalidate the safety analyses. ~~Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.~~

INSERT 3

SR 3.1.8.4

The SDM is verified by comparing the RCS boron concentration to a SHUTDOWN MARGIN requirement curve that was generated by taking into account estimated RCS boron concentrations, core power defect, control bank position, RCS average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and isothermal temperature coefficient (ITC).

~~The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.~~

INSERT 3

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.
4. UFSAR, Section 14.6.
5. Letter from R. W. Kober (RGE) to T. E. Murley (NRC), Subject: "Startup Reports," dated July 9, 1984.
6. Letter from J. P. Durr (NRC) to B. A. Snow (RGE), Subject: "Inspection Report No. 50-244/88-06," dated April 28, 1988.

SR 3.2.1.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)$ 1.0815 (Ref. 4). $F_Q^C(Z)$is then compared to its specified limits.

The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

~~The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).~~

INSERT 3

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power above 50% RTP and directly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 61 core elevations. $F_Q^W(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 8% inclusive and

b. Upper core region, from 92 to 100% inclusive.

The top and bottom 8% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated, an evaluation of the expression below is required to account for any increase to $F_Q^M(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression maximum over z $[F_Q^C(Z) \ / \ K(Z)]$, it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels $\geq$ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

~~The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.~~

~~The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.~~

INSERT 3

REFERENCES

1. 10 CFR 50.46.

2. UFSAR 15.4.5.4.3

3. Atomic Industrial Forum (AIF) GDC-29, Issued for comment July 10, 1967

4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

5. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control (and) F_Q Surveillance Technical Specification," February 1994.

A.3

Reduction in the Overpower ΔT and Overtemperature ΔT trip setpoints by $\geq$ 1% for each 1% by which $F^N_{\Delta H}$ exceeds its limit, ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.4

Verification that $F^N_{\Delta H}$ has been restored within its limit by performing SR 3.2.2.1 or SR 3.2.2.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1 ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If the Required Actions of A.1 through A.4 cannot be met within their associated Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

The allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

After each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

INSERT 3

~~The Frequency of 31 EFPD is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.~~ When the plant is already performing SR 3.2.2.2 to satisfy other requirements, SR 3.2.2.2 does not need to be suspended in order to perform SR 3.2.2.1 since the performance of SR 3.2.2.2 meets the requirements of SR 3.2.2.1.

SR 3.2.2.2

During power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables.

With an NIS power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. ~~Performing SR 3.2.2.2 at a Frequency of 24 hours provides an accurate alternative means for ensuring that $F^N_{\Delta H}$ remains within limits and the core power distribution is consistent with the safety analyses. A Frequency of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map.~~ INSERT 3

This Surveillance is modified by a Note, which states that it is required only when one power range channel is inoperable and the THERMAL POWER is ≥ 75% RTP.

REFERENCES

1. 10 CFR 50.46.

2. UFSAR, Section 15.4.5.1.

3. Atomic Industrial Forum (AIF) GDC 29, Issued for comment July 10 1967.

4. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condtion for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits. ~~The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.~~ INSERT 3

REFERENCES

1. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specification", February 1994.
2. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
3. UFSAR, Section 7.7.2.6.4.

assumptions, Required Action A.6 requires verification that $F_Q(Z)$ as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F^N_{\Delta H}$ are within their specified limits within 24 hours after reaching equilibrium condition at RTP. As an added precaution, if the core power does not reach equilibrium condition at RTP within 24 hours, but it increases slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to eliminate the indicated tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are adjusted to eliminate the indicated tilt and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the plant must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to ≤ 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. ~~The Frequency of 7 days takes into account other information and alarms available in the control room~~ INSERT 3

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and one power range channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

For those causes of quadrant power tilt that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of the core power tilt.

SR 3.2.4.2

This surveillance is modified by a Note, which states that it is not required until 24 hours after the input from one or more Power Range Neutron Flux channel is inoperable and the THERMAL POWER is > 75% RTP. With the input from a NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased.

When one NIS power range channel input is inoperable and THERMAL POWER is > 75% RTP, a full core flux map should be performed to verify the core power distribution ~~instead of using the three OPERABLE power range channel inputs to verify QPTR by performing SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1, at a Frequency of 24 hours~~. Performing a full core flux map provides an accurate alternative means for ensuring that $F_Q(Z)$ and $F^N_{\Delta H}$ remain within limits and the core power distribution is consistent with the safety analysis. INSERT 3

REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Section 15.4.5.
3. Atomic Industrial Forum (AIF) GDC 29, Issued for comment July 10, 1967.

X.1 and X.2

If the Required Action and Associated Completion Time of Condition W is not met, the plant must be placed in a MODE where the Functions are no longer required. To achieve this status, action be must initiated immediately to fully insert all rods and the CRD System must be incapable of rod withdrawal within 1 hour. These Completion Times are reasonable, based on operating experience to exit the MODE of Applicability in an orderly manner.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel 1, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel 2, Channel 3, and Channel 4 (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies (Ref. 8).

SR 3.3.1.1

A CHANNEL CHECK is required for the following RTS trip functions:

- Power Range Neutron Flux-High;
- Power Range Neutron Flux-Low;
- Intermediate Range Neutron Flux;
- Source Range Neutron Flux;
- Overtemperature ΔT;
- Overpower ΔT;
- Pressurizer Pressure-Low;
- Pressurizer Pressure-High;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (Single Loop);

- Reactor Coolant Flow-Low (Two Loops); and
- SG Water Level-Low Low

Performance of the CHANNEL CHECK ~~onceevery 12 hours~~ ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Channel check acceptance criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

~~The Frequency of 12 hours is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~

INSERT 3

SR 3.3.1.2

This SR compares the calorimetric heat balance calculation to the NIS Power Range Neutron Flux-High channel output ~~every 24 hours~~. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is still OPERABLE but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is then declared inoperable.

This SR is modified by a Note which states that this Surveillance is required to be performed within 12 hours after power is ≥ 50% RTP. At lower power levels, calorimetric data are inaccurate.

~~The Frequency of every 24 hours is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period~~.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

INSERT 3

SR 3.3.1.3

This SR compares the incore system to the NIS channel output ~~every 31 effective full power days (EFPD)~~. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is then declared inoperable. This surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

This SR is modified by two Notes. Note 1 clarifies that the Surveillance is required to be performed within 7 days after THERMAL POWER is≥ 50% RTP but prior to exceeding 90% RTP following each refueling and if it has not been performed within the last 31 EFPD. Note 2 states that performance of SR 3.3.1.6 satisfies this SR since it is a more comprehensive test.

~~The Frequency of every 31 EFPD is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.~~

INSERT 3

SR 3.3.1.4

This SR is the performance of a TADOT ~~every 31 days on a STAGGERED TEST BASIS~~ of the RTB, and the RTB Undervoltage and Shunt Trip Mechanisms. This test shall verify OPERABILITY by actuation of the end devices.

The test shall include separate verification of the undervoltage and shunt trip mechanisms except for the bypass breakers which do not require separate verification since no capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.11. However, the bypass breaker test shall include a local shunt trip. This test must be performed on the bypass breaker prior to placing it in service to take the place of a RTB.

~~The Frequency of every 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data.~~

INSERT 3

SR 3.3.1.5

This SR is the performance of an ACTUATION LOGIC TEST on the RTS Automatic Trip Logic ~~every 31 days on a STAGGERED TEST BASIS~~. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. All possible logic combinations, with and without applicable permissives, are tested for each protection function. ~~The Frequency of every 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data.~~

INSERT 3

SR 3.3.1.6

This SR is a calibration of the excore channels to the incore channels ~~every 92 EFPD~~. If the measurements do not agree, the excore channels are still OPERABLE but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are then declared inoperable. This surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

A minimum of 2 thimbles per quadrant and sufficient movable incore detectors shall be operable during recalibration of the excore axial off-set detection system. To calibrate the excore detector channels, it is only necessary that the movable incore system be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.

This SR has been modified by a Note stating that this Surveillance is required to be performed within 7 days after THERMAL POWER is≥ 50% RTP but prior to exceeding 90% RTP following each refueling.

~~The Frequency of 92 EFPD is adequate based on industry operating experience, considering instrument reliability and operating history data for instrument drift.~~

INSERT 3

SR 3.3.1.7

This SR is the performance of a COT ~~every 92 days~~ for the following RTS functions:

- Power Range Neutron Flux-High;
- Source Range Neutron Flux (in MODE 3, 4, or 5 with CRD System capable of rod withdrawal or all rods not fully inserted);
- Overtemperature ΔT;
- Overpower ΔT;
- Pressurizer Pressure-Low;

- Pressurizer Pressurizer-High;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (Single Loop);
- Reactor Coolant Flow-Low (Two Loops); and
- SG Water Level-Low Low

A COT is performed on each required channel to ensure the channel will perform the intended Function. The as-found setpoints must be within the COT Acceptance Criteria specified within plant procesures. The as-left values must be consistent with the setting tolerance used in the setpoint methodology (Ref. 8).

This SR is modified by a Note that provides a 4 hour delay in the requirement to perform this surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the plant is in MODE 3 with the RTBs closed for greater than 4 hours, this SR must be performed within 4 hours after entry into MODE 3.

INSERT 3

~~The Frequency of 92 days is consistent with Reference 9.~~

SR 3.3.1.8

This SR is the performance of a COT as described in SR 3.3.1.7 for the Power Range Neutron Flux-Low, Intermediate Range Neutron Flux, and Source Range Neutron Flux (MODE 2), except that this test also includes verification that the P-6 and P-10 interlocks are in their required state for the existing plant condition. This SR is modified by two Notes that provide a 4 hour delay in the requirement to perform this surveillance. These Notes allow a normal shutdown to be completed and the plant removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency INSERT 1 ~~of every 92 days~~ applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and 4 hours after reducing power below P-10 or P-6.

The MODE of Applicability for this surveillance is < 6% RTP for the power range low and intermediate range channels and < 5E-11amps for the Source range channels. Once the plant is in MODE 3, this surveillance is no longer required. If power is to be maintained < 6% RTP or < 5E-11amps for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit, unless performed INSERT 1 ~~within the prior 92 days.~~ Four hours is a reasonable time to complete the required testing or place the plant in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical or after reducing power into the applicable MODE (< 6% RTP or < 5E-11amps) for periods > 4 hours. INSERT 3

SR 3.3.1.9

This SR is the performance of a TADOT for the Undervoltage-Bus 11A and 11B and Underfrequency-Bus 11A and 11B trip Functions. ~~The Frequency of every 92 days is consistent with Reference 9.~~

This SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to Bus 11A and 11B undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION required by SR 3.3.1.10. INSERT 3

SR 3.3.1.10

This SR is the performance of a CHANNEL CALIBRATION for the following RTS Functions:

- Power Range Neutron Flux-High;
- Power Range Neutron Flux-Low;
- Intermediate Range Neutron Flux;
- Source Range Neutron Flux;
- Overtemperature ΔT;
- Overpower ΔT;
- Pressurizer Pressure-Low;
- Pressurizer Pressure-High;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (Single Loop);

- Reactor Coolant Flow-Low (Two Loops);
- Undervoltage-Bus 11A and 11B;
- Underfrequency-Bus 11A and 11B;
- SG Water Level-Low Low;
- Turbine Trip-Low Autostop Oil Pressure; and
- Reactor Trip System Interlocks.

~~A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling.~~ CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the plant specific setpoint methodology (Ref. 8). The difference between the current as-found values and the previous test as-left values must be consistent with the drift allowance used in the setpoint methodology.

~~The Frequency of 24 months is based on the assumption of 24 month calibration intervals in the determination of the magnitude of equipment drift in the setpoint methodology.~~

With respect to RTDs, whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD) sensors shall include an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. This is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 50% RTP. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the plant must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. ~~The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant~~

~~outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 24 month Frequency.~~ INSERT 3

SR 3.3.1.11

This SR is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, and the SI Input from ESFAS trip Functions. ~~This TADOT is performed every 24 months.~~ This test independently verifies the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers.

~~The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.~~ INSERT 3

SR 3.3.1.12

This SR is the performance of a TADOT for Turbine Trip Functions which is performed prior to reactor startup if it has not been performed within the last 31 days. This test shall verify OPERABILITY by actuation of the end devices.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

This SR is modified by a Note stating that verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical because portions of this test cannot be performed with the reactor at power.

SR 3.3.1.13

~~This SR is the performance of a COT of the RTS interlocks every 24 months.~~

~~The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.~~

INSERT 3

REFERENCES

1. Atomic Industry Forum (AIF) GDC 14, Issued for comment July 10, 1967.
2. 10 CFR 50.67.
3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
4. UFSAR, Chapter 7.
5. UFSAR, Chapter 6.
6. UFSAR, Chapter 15.
7. IEEE-279-1971.
8. EP-3-S-0505, "Instrument Setpoint/Loop Accuracy Calculation Methodology".
9. ~~WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.~~

Deleted

SURVEILLANCE REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1. Each channel of process protection supplies both trains of the ESFAS. When testing Channel 1, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel 2, Channel 3, and Channel 4 (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

SR 3.3.2.1

This SR is the performance of a CHANNEL CHECK for the following ESFAS Functions:

- SI-Containment Pressure-High;
- SI-Pressurizer Pressure-Low;
- SI-Steam Line Pressure-Low;
- CS-Containment Pressure-High High;
- Steam Line Isolation-Containment Pressure-High High;
- Steam Line Isolation-High Steam Flow Coincident with SI and T_{avg}-Low;
- Steam Line Isolation-High-High Steam Flow Coincident with SI;
- Feedwater Isolation-SG Water Level-High; and
- AFW-SG Water Level-Low Low.

Performance of the CHANNEL CHECK~~onceevery 12 hours~~ ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

CHANNEL CHECK acceptance criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

~~The Frequency of 12 hours is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~ INSERT 3

SR 3.3.2.2

This SR is the performance of a COT ~~every 92 days~~ for the following ESFAS functions:

- SI-Containment Pressure-High;
- SI-Pressurizer Pressure-Low;
- SI-Steam Line Pressure-Low;
- CS-Containment Pressure-High High;
- Steam Line Isolation-Containment Pressure-High High;
- Steam Line Isolation-High Steam Flow Coincident with SI and T_{avg}-Low;
- Steam Line Isolation-High-High Steam Flow Coincident with SI;
- Feedwater Isolation-SG Water Level-High; and
- AFW-SG Water Level-Low Low.

A COT is performed on each required channel to ensure the channel will perform the intended Function. Setpoints must be found to be within the COT Acceptance Criteria specified in plant procedures. The as-left values must be consistent with the drift allowance used in the setpoint methodology.

~~The Frequency of 92 days is consistent with in Reference 7. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.~~ INSERT 3

SR 3.3.2.3

This SR is the performance of a TADOT ~~every 92 days~~. This test is a check of the AFW-Undervoltage-Bus 11A and 11B Function.

The test includes trip devices that provide actuation signals directly to the protection system. The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION. ~~The Frequency of 92 days is adequate based on industry operating experience, considering instrument reliability and operating history data.~~

INSERT 3

SR 3.3.2.4

This SR is the performance of a TADOT ~~every 24 months~~. This test is a check of the SI, CS, Containment Isolation, Steam Line Isolation, and AFW Manual Initiations, and the AFW-Trip of Both MFW Pumps Functions. Each Function is tested up to, and including, the master transfer relay coils. ~~The Frequency of 24 months is based on industry operating experience and is consistent with the typical refueling cycle.~~

INSERT 3

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Manual Initiations, and AFW-Trip of Both MFW Pumps Functions have no associated setpoints.

SR 3.3.2.5

This SR is the performance of a CHANNEL CALIBRATION ~~every 24 months~~ of the following ESFAS Functions:

- SI-Containment Pressure-High;
- SI-Pressurizer Pressure-Low;
- SI-Steam Line Pressure-Low;
- CS-Containment Pressure-High High;
- Steam Line Isolation-Containment Pressure-High High;
- Steam Line Isolation-High Steam Flow Coincident with SI and T_{avg}-Low;
- Steam Line Isolation-High-High Steam Flow Coincident with SI;
- Feedwater Isolation-SG Water Level-High;
- AFW-SG Water Level-Low Low; and
- AFW-Undervoltage-Bus 11A and 11B.

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the plant specific setpoint methodology. The "as left" values must be consistent with the drift allowance used in the setpoint methodology.

~~The Frequency of 24 months is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.~~ INSERT 3

SR 3.3.2.6

This SR ensures the SI-Pressurizer Pressure-Low and SI-Steam Line Pressure-Low Functions are not bypassed when pressurizer pressure > 2000 psig while in MODES 1, 2, and 3. Periodic testing of the pressurizer pressure channels is required to verify the setpoint to be less than or equal to the limit.

The difference between the current as-found values and the previous test as-left values must be consistent with the drift allowance used in the setpoint methodology (Ref. 6). The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.

If the pressurizer pressure interlock setpoint is nonconservative, then the Pressurizer Pressure-Low and Steam Line Pressure-Low Functions are considered inoperable. Alternatively, the pressurizer pressure interlock can be placed in the conservative condition (nonbypassed). If placed in the nonbypassed condition, the SR is met and the Pressurizer Pressure-Low and Steam Line Pressure-Low Functions would not be considered inoperable. INSERT 3

SR 3.3.2.7

This SR is the performance of an ACTUATION LOGIC TEST on all ESFAS Automatic Actuation Logic and Actuation Relays Functions ~~every 24 months~~. This test includes the application of various simulated or actual input combinations in conjunction with each possible interlock state and verification of the required logic output. Relay and contact operation is verified by a continuance check or actuation of the end device.

~~The Frequency of 24 months is based on operating experience and the need to perform this testing during a plant shutdown to prevent a reactor trip from occurring.~~ INSERT 3

G.1

If one channel for Function 7 or 10 cannot be restored to OPERABLE status within the required Completion Time of Condition D, the plant must take immediate action to prepare and submit a special report to the NRC. This report shall be submitted within the following 14 days from the time the action is required. This report discusses the alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation, the degree to which the alternate means are equivalent to the installed PAM channels, the areas in which they are not equivalent, and a schedule for restoring the normal PAM channels.

These alternate means must have been developed and tested and may be temporarily installed if the normal PAM channel(s) cannot be restored to OPERABLE status within the allotted time.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SR 3.3.3.1

Performance of the CHANNEL CHECK ~~once every 31 days~~ ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar plant instruments located throughout the plant.

Channel check acceptance criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

~~The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during~~

~~normal operational use of the displays associated with the LCO required channels.~~ INSERT 3

SR 3.3.3.2

~~A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling.~~ CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to the measured parameter with the necessary range and accuracy. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors shall include an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. This is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. ~~The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.~~ INSERT 3

REFERENCES

1. UFSAR, Section 7.5.2.
2. Regulatory Guide 1.97, Rev. 3.
3. NUREG-0737, Supplement 1, "TMI Action Items."
4. UFSAR, Section 6.2.5.

significantly reduce the probability that the LOP DG start instrumentation will trip when necessary.

SR 3.3.4.1

This SR is the performance of a TADOT ~~every 31 days~~. This test checks trip devices that provide actuation signals directly. For these tests, the relay trip setpoints are verified and adjusted as necessary to ensure the LSSS can still be met. ~~The 31 day Frequency is based on the known reliability of the relays and controls and has been shown to be acceptable through operating experience.~~ INSERT 3

SR 3.3.4.2

This SR is the performance of a CHANNEL CALIBRATION ~~every 24 months, or approximately at every refueling,~~ of the LOP DG start instrumentation for each 480 V bus.

The voltage setpoint verification, as well as the time response to a loss of voltage and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay.

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

~~The Frequency of 24 months is based on operating experience consistent with the typical industry refueling cycle and is justified by the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.~~ INSERT 3

REFERENCES

1. UFSAR, Section 8.3.
2. UFSAR, Chapter 15.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.5-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.5.1

Performance of the CHANNEL CHECK ~~onceevery 24 hours~~ ensures that a gross failure of instrumentation has not occurred and the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The CHANNEL CHECK agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

~~The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~ INSERT 3

SR 3.3.5.2

A COT is performed ~~every 92 days~~ on each required channel to ensure the channel will perform the intended Function. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). This test verifies the capability of the instrumentation to provide the containment ventilation system isolation. The setpoint shall be left consistent with the current plant specific calibration procedure tolerance. INSERT 3

SR 3.3.5.3

This SR is the performance of an ACTUATION LOGIC TEST. All possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay is tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. ~~This test is performed every 24 months. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.~~ INSERT 3

SR 3.3.5.4

~~A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling.~~ CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

~~The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.~~

INSERT 3

REFERENCES

1. 10 CFR 50.67.

2. NUREG-1366.

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time of Condition A or B has not been met and the plant is in MODE 1, 2, 3, or 4. The plant must be brought to a MODE that minimizes accident risk. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1 and D.2

Condition D applies when the Required Action and associated Completion Time of Condition A or B has not been met during movement of irradiated fuel assemblies. Movement of irradiated fuel assemblies must be suspended immediately to reduce the risk of accidents that would require CREATS actuation. This places the plant in a condition that minimizes risk. This does not preclude movement of fuel or other components to a safe position.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which CREATS Actuation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK ~~onceevery 12 hours~~ ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

CHANNEL CHECK acceptance criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

~~The Frequency of 12 hours is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during~~

~~normal operational use of the displays associated with the LCO required channels.~~ INSERT 3

SR 3.3.6.2

This SR is the performance of a COT ~~once every 92 days~~ on each required channel to ensure the channel will perform the intended function. This test verifies the capability of the instrumentation to provide the automatic CREATS actuation. The setpoints shall be left consistent with the plant specific calibration procedure tolerance. ~~The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.~~ INSERT 3

SR 3.3.6.3

This SR is the performance of a TADOT of the Manual Initiation Function ~~every 24 months~~. The Manual Initiation Function is tested up to, and including, the master relay coils.

~~The Frequency of 24 months is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.~~ INSERT 3

The SR is modified by a Note that excludes verification of setpoints because the Manual Initiation Function has no setpoints.

SR 3.3.6.4

~~This SR is the performance of a CHANNEL CALIBRATION every 24 months, or approximately at every refueling.~~ CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

~~The Frequency of 24 months is based on operating experience and is consistent with the typical industry refueling cycle.~~ INSERT

SR 3.3.6.5

This SR is the performance of an ACTUATION LOGIC TEST. All possible logic combinations are tested for the CREATS actuation instrumentation. In addition, the master relay is tested forcontinuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is acceptable based on instrument reliability and operating experience. INSERT 3

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to determine the cause for the off normal condition, to adjust plant parameters, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

~~Since Required Action A.1 allows a Compldion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.~~

INSERT 3

SR 3.4.1.2

~~Since Required Action A.1 allows a Compldion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.~~

INSERT 3

SR 3.4.1.3

Measurement of RCS total flow rate ~~once every 24 months~~ verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. This verification may be performed via a precision calorimetric heat balance or other accepted means.

INSERT 3

~~The Frequency of 24 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.~~ Verification of RCS flow rate on a shorter interval is not required since this parameter is not expected to vary during steady state operation as there are no RCS loop isolation valves or other installed devices which could significantly alter flow. Reduced performance of a reactor coolant pump (RCP) would be observable due to bus voltage and frequency changes, and installed alarms that would result in operator investigation.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the plant in the best condition for performing the SR. The Note states that the SR shall be performed within 7 days after reaching 95% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 95% RTP to obtain the stated RCS flow accuracies.

REFERENCES

1. UFSAR, Chapter 15.

2. NRC Memorandum from E.L. Jordan, Assistant Director for Technical Programs, Division of Reactor Operations Inspection to Distribution; Subject: "Discussion of Licensed Power Level (AITS F14580H2)," dated August 22, 1980.

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with $K_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period due to the proximity to MODE 2 conditions. The allowed time is reasonable, based on operating experience, to reach MODE 2 with $K_{eff} < 1.0$ in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

This SR verifies that RCS T_{avg} in each loop is ≥ 540°F within 30 minutes prior to achieving criticality. This ensures that the minimum temperature for criticality is being maintained just before criticality is reached. The 30 minute time period is long enough to allow the operator to adjust temperatures or delay criticality so the LCO will not be violated, thereby providing assurance that the safety analyses are not violated.

SR 3.4.2.2

RCS loop average temperature is required to be verified at or above 540°F every 30 minutes in MODE 1, and in MODE 2 with $k_{eff} \geq 1.0$. The 30 minute frequency is sufficient based on the low likelihood of large temperature swings without the operators knowledge.

INSERT 3

This SR is modified by a Note that only requires the SR to be performed if any RCS loop T_{avg} is < 547°F and the low T_{avg} alarm is either inoperable or not reset. The T_{avg} alarm provides operator indication of low RCS temperature without requiring independent verification while a T_{avg} > 547°F in both RCS loops is within the accident analysis assumptions. If the T_{avg} alarm is to be used for this SR, it should be calibrated consistent with industry standards.

This surveillance is replaced by SR 3.1.8.2 during PHYSICS TESTING.

REFERENCES

1. None.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required ~~every 30 minutes~~ when RCS pressure and temperature conditions are undergoing planned changes. ~~This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.~~ INSERT 3

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. WCAP-14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 1, December 1994.
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. ASTM E 185-82, July 1982.
5. 10 CFR 50, Appendix H.
6. Regulatory Guide 1.99, Revision 2, May 1988.
7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

Operation in other MODES is covered by:

LCO 3.4.5,	"RCS Loops - MODES 1 ≤ 8.5% RTP, 2, AND 3";
LCO 3.4.6,	"RCS Loops - MODE 4";
LCO 3.4.7,	"RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8,	"RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.4,	"Residual Heat Removal (RHR) and Coolant Circulation-Water Level ≥ 23 Ft" (MODE 6); and
LCO 3.9.5,	"Residual Heat Removal (RHR) and Coolant Circulation-Water Level < 23 Ft" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 1 < 8.5% RTP. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 1 < 8.5% RTP from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification ~~every 12 hours~~ that each RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. Use of control board indication for these parameters is an acceptable verification. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.~~

INSERT 3

B.1

If restoration of the inoperable loop is not possible within 72 hours, the plant must be brought to MODE 4. In MODE 4, the plant may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1, C.2, and C.3

If two RCS loops are inoperable, or no RCS loop is in operation, except during conditions permitted by the Note in the LCO section, all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets. All operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification ~~every 12 hours~~ that the required RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. Use of the control board indication for these parameters is an acceptable verification. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.~~ INSERT 3

SR 3.4.5.2

This SR requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 16% for two RCS loops. If the SG secondary side narrow range water level is < 16%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of reactor or decay heat. ~~The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.~~ INSERT 3

SR 3.4.5.3

Verification that the required RCP is OPERABLE ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump that is not in operation. ~~The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ INSERT 3

REFERENCES

1. UFSAR Section 15.1.5.
2. UFSAR Section 15.4.3.
3. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic XV-9, Startup of an Inactive Loop, R. E. Ginna," dated August 26, 1981.
4. UFSAR Sections 14.6.1.5.6 and 15.2.5.
5. UFSAR Section 14.6.1.5.5.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification ~~every 12 hours~~ that one RCS or RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. Use of control board indication for these parameters is an acceptable verification. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.~~ INSERT 3

SR 3.4.6.2

This SR requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 16%. If the SG secondary side narrow range water level is < 16%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. ~~The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.~~ INSERT 3

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump that is not in operation. ~~The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ INSERT 3

REFERENCES

1. UFSAR, Section 14.6.1.2.6.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification ~~every 12 hours~~ that one RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. Use of control board indication for these parameters is an acceptable verification. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.~~

INSERT 3

SR 3.4.7.2

This SR requires verification of SG OPERABILITY. Verifying that at least one SG is OPERABLE by ensuring its secondary side narrow range water level is ≥ 16% ensures an alternate decay heat removal method in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. ~~The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.~~

INSERT 3

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the standby RHR pump. If secondary side water level is ≥ 16% in at least one SG, this Surveillance is not needed. ~~The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~

INSERT 3

REFERENCES

1. UFSAR, Section 14.6.1.2.6
2. NRC Information Notice 95-35

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification~~every 12 hours~~that one RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.~~ INSERT 3

SR 3.4.8.2

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the standby pump. ~~The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ INSERT 3

REFERENCES

1. None.

B.1 and B.2

If the pressurizer heaters capacity is < 100 KW, the ability to maintain RCS pressure to support natural circulation may no longer exist. By maintaining RCS pressure control, a margin to subcooling is provided. The value of 100 KW is based on the amount needed to support natural circulation after accounting for heat losses through the pressurizer insulation during an extended loss of offsite power event.

If the capacity of the pressurizer heaters is not within the limit, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. ~~The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumptions.~~ Alarms are also available for early detection of abnormal level indications. INSERT 3

SR 3.4.9.2

This SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power required. This may be done by testing the power supply output by verifying the electrical load on Buses 14 and 16 with the respective heater groups on and off. ~~The Frequency of 92 days is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.~~ INSERT 3

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

INSERT 3

Block valve cycling verifies that the valve(s) can be closed if needed. ~~The basis for the Frequency of 92 days is the ASME Code (Ref. 2).~~ If the block valve is closed to isolate a PORV that is OPERABLE and is not leaking in excess of the limits of LCO 3.4.13, "RCS Operational LEAKAGE," then opening the block valve is necessary to verify that the PORV can be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency ~~of 92 days~~. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Actions fulfills the SR).

The Note modifies this SR by stating that it is not required to be performed with the block valve closed per LCO 3.4.13. This prevents the need to open the block valve when the associated PORV is leaking > 10 gpm creating the potential for a plant transient.

SR 3.4.11.2

This SR requires a complete cycle of each PORV using the nitrogen accumulators. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. ~~The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice.~~

INSERT 3

REFERENCES

1. UFSAR, Section 15.2.
2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

disabling of a charging pump is necessary since RV 203 cannot mitigate a charging/letdown mismatch event if RHR is providing decay heat removal above MODE 5 and three charging pumps are operating.

The passive vent must be sized ≥ 1.1 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel and to protect the RHR system from overpressurization.

The Completion Time of 8 hours to depressurize the RCS and establish a vent considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, all SI pumps must be verified incapable of injecting into the RCS when the PORVs provide the RCS vent path (LCO 3.4.12.a) and a minimum of two SI pumps must be verified incapable of injecting into the RCS when the RCS is depressurized and an RCS vent ≥ 1.1 square inches is established (LCO 3.4.12.b). The SI pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the following:

a. placing the pump control switch in the pull-stop position and closing at least one valve in the discharge flow path;

b. locking closed a manual isolation valve in the injection path; or

c. closing a motor operated isolation valve in the injection path and removing the AC power source.

The flowpaths through the test connections associated with the ECCS accumulator check valves (i.e., lines containing air operated valves 839A, 839B, 840A, and 840B) and the ECCS accumulator fill lines (i.e., lines containing air operated valves 835A and 835B) do not have to be isolated for this SR since the potential mass addition from a single SI pump through these six lines is limited by the installed orifices to less than that assumed for the charging/letdown mismatch analysis.

The ECCS accumulator motor operated isolation valves can be verified closed by use of control board indication for valve position. This verification is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. If the accumulator pressure is less than this limit, no verification is required since the accumulator cannot pressurize the RCS to or above the PORV setpoint.

~~The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment. The Frequency of every 12 hours thereafter for SR 3.4.12.3 ensures that the ECCS accumulator motor operated isolation valves are maintained closed and do not result in a potential LTOP actuation.~~ INSERT 3

SR 3.4.12.2

See SR 3.4.12.1

SR 3.4.12.3

See SR 3.4.12.1

SR 3.4.12.4

The RCS vent of ≥ 1.1 square inches is proven OPERABLE by verifying its open condition ~~either:~~

~~a. Once every 12 hours for a vent (e.g., valve) that cannot be locked.~~

~~b. Once every 31 days for a vent (e.g., valve) that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.~~ INSERT 3

The passive vent arrangement must be ≥ 1.1 square inches and be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12.b.

SR 3.4.12.5

The PORV block valve must be verified open ~~every 72 hours~~ to provide the flow path for each required PORV to perform its function when actuated. The valve may be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required to be removed, and the manual operator is not required to be locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops

excessive leakage or does not close (sticks open) after relieving an overpressure situation.

~~The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.~~ INSERT 3

SR 3.4.12.6

Performance of a CHANNEL OPERATIONAL TEST (COT) is required ~~every 31 days~~ on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is therefore not required.

A Note has been added indicating that this SR is required to be performed within 12 hours after decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature specified in the PTLR if it has not been performed ~~within the previous 31 days~~. Depending on the cooldown rate, the COT may not have been performed before entry into the LTOP MODES. The test must be performed within 12 hours after entering the LTOP MODES. The 12 hours considers the unlikelihood of a low temperature overpressure event during this time.

SR 3.4.12.7 INSERT 1 INSERT 1

Verification once within 12 hours and ~~every 31 days~~ thereafter that power is removed from each ECCS accumulator motor operated isolation valve ensures that at least two independent actions must occur before the accumulator is capable of injecting into the RCS. ~~Since power is removed under administrative control and valve position is verified every 12 hours, the performance of this surveillance once within 12 hours and every 31 days thereafter will provide assurance that power is removed~~.

This SR is modified by a Note which states that the Surveillance is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed in the PTLR. If the accumulator pressure is below this limit, the LTOP limit cannot be exceeded and the surveillance is not required. INSERT 3

SR 3.4.12.8

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required ~~every 24 months~~ to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input. INSERT 3

REFERENCES 1. 10 CFR 50, Appendix G.

Deleted

2. ~~Generic Letter 88-11, "NRC Position on Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations."~~

3. UFSAR, Section 5.2.2.

4. 10 CFR 50, Section 50.46.

5. 10 CFR 50, Appendix K.

6. Letter from D. L. Ziemann, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment No. 28 to Provisional Operating License No. DPR-18," dated July 26, 1979.

7. Generic Letter 90-06, "Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any RCS pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limits, or if the Required Action of Condition A cannot be completed within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE which is not allowed by this LCO, would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides

sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and volume control tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

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

~~The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.~~ INSERT 3

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

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

~~The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation~~

~~monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Reference 5).~~ INSERT 3

REFERENCES

1. Atomic Industry Forum (AIF) GDC 16, Issued for comment July 10, 1967.
2. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."
3. UFSAR, Chapter 15.
4. NEI 97-06, Steam Generator Program Guidelines
5. ~~EPRI, Pressurized Water Reactor Primary-to-Secondary Leak Guidelines~~ Deleted

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation. The use of a valve other than the previously leaking PIV must include consideration that the plant may no longer be in an analyzed condition. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

B.1 and B.2

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage due to reduced RCS pressure while reducing the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpmper inch of nominal valve diameter up to 5 gpm maximum applies to each valve and should be based on an RCS pressure of $\pm$ 20 psig of normal system operating pressure. Leakage testing requires a stable pressure condition.

For multiple in-series PIVs, the leakage requirement applies to each valve individually, except as noted below, and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other in-series valve meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

The SI hot leg injection lines are each configured with two check valves and a motor operated valve in series. Each of these components independently is considered a qualified pressure boundary. The two check valves function as a single pressure isolation barrier and the motor operated valve serves as the second pressure isolation barrier to prevent an intersystem LOCA. Both barriers need to be tested. Testing of the check valves (877A, 877B, 878F, and 878H) and the motor operated valves (878A and 878C) identified as PIVs in the SI hot leg injection lines is to be performed at least once every 40 months. This surveillance interval is allowed since the two SI hot leg injection lines are maintained closed to address pressurized thermal shock (PTS) concerns (Ref. 7 and Ref. 11).

~~Testing of the RCS PIVs in the SI cold leg injection lines and RHR system is to be performed every 24 months, a typical refueling cycle. The 24 month Frequency is consistent with 10 CFR 50.55a(f) (Ref. 10) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code, (Ref. 9), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~ INSERT 3

In addition to the periodic testing requirements, testing must be performed once after the valve has been opened by flow, exercised, or had maintenance performed on it to ensure tight reseating. This maintenance does not include minor activities such as packing adjustments which do not affect the leak tightness of the valve. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. A limit of 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance.

<u>SR 3.4.14.2</u>

See SR 3.4.14.1

REFERENCES

1. 10 CFR 50.2.

2. 10 CFR 50.55a(c).

3. Atomic Industry Forum (AIF) GDC 53, Issued for comment July 10, 1967.

4. WASH-1400 (NUREG-75/014), "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Appendix V, October 1975.

5. NUREG-0677, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," May 1980.

6. Generic Letter, "LWR Primary Coolant System Pressure Isolation Valves," dated February 23, 1980.

7. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," and associated SER on Primary Coolant System Pressure Isolation Valves (WASH-1400, Event V), dated April 20, 1981. (ML010542030)

8. EG&G Report, EGG-NTAP-6175.

9. ~~ASME Code for Operation and Maintenance of Nuclear Power Plants.~~ Deleted

10. ~~10 CFR 50.55a(f).~~ Deleted

11. Letter from D. M. Crutchfield, NRC, to J.E. Maier, RGE, Subject: "TMI-2 Category "A" Items" and associated SER for Amendment No. 42 to Provisional Operating License No. DPR-18, dated May 11, 1981. (ML010540356)

Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy period of time.

E.1 and E.2

If a Required Action of Condition A, C, or D cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

This SR requires the performance of a CHANNEL CHECK of the containment atmosphere radioactivity monitors. The check gives reasonable confidence that the channels are operating properly. ~~The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.~~ INSERT 3

SR 3.4.15.2

This SR requires the performance of a CHANNEL OPERATIONAL TEST (COT) on the containment atmosphere radioactivity monitors. The test ensures that the monitors can perform their function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. ~~The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.~~ INSERT 3

SR 3.4.15.3

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. ~~The Frequency of 24 months considers channel reliability and operating experience has proven that this Frequency is acceptable.~~ INSERT 3

SR 3.4.15.4

See SR 3.4.15.3

REFERENCES

1. Atomic Industry Forum (AIF) GDC 16 and 34, Issued for comment July 10, 1967.
2. Regulatory Guide 1.45.
3. IE Bulletin No. 80-24, "Prevention of Damage Due to Water Leakage Inside Containment."
4. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," 1981.
5. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."
6. Letter from D. C. Dilanni, NRC, to R. W. Kober, RG&E, Subject: "Generic Letter 84-04," dated September 9, 1986.
7. NUREG-0821, "Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Nuclear Power Plant," December 1982.
8. Letter from Guy S. Vissing (NRC) to Robert C. Mecredy (RG&E), "Staff Review of the Submittal by Rochester Gas and Electric Company to Apply Leak-Before-Break Status to Portions of the R.E. Ginna Nuclear Power Plant Residual Heat Removal System Piping", dated February 25, 1999.

C.1

If the gross specific activity is not within limit, the change within 8 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents automatically venting the SG to the environment in an SGTR event. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

This SR requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant ~~at least once every 7 days~~. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with $T_{avg} \geq 500°F$. ~~The 7 day Frequency considers the unlikelihood of a gross fuel failure during this time.~~ INSERT 3

SR 3.4.16.2

This SR is only performed in MODE 1 to ensure iodine remains within limits during normal operation and following fast power changes when fuel failure is more likely to occur. ~~The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days.~~ INSERT 3 The Frequency, between 2 and 10 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours and ~~every 184 days (6 months)~~ INSERT 1 thereafter. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The

analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. ~~The Frequency recognizes $\bar{E}$ does not change rapidly.~~ INSERT 3

This SR is modified by a Note that indicates sampling is only required to be performed in MODE 1 such that equilibrium conditions are present during the sample.

REFERENCES

1. 10 CFR 50.67.

2. Design Analysis DA-NS-2001-084, Steam Generator Tube Rupture Offsite and Control Room Doses.

power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status is justified in WCAP-15049-A, Rev. 1 (Ref. 10).

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and pressurizer pressure reduced to ≤ 1600 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If both accumulators are inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator motor-operated isolation valve shall be verified to be fully open ~~every 12 hours~~. Use of control board indication for valve position is an acceptable verification. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. ~~This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.~~

INSERT 3

SR 3.5.1.2

The borated water volume and nitrogen cover pressure shall be verified ~~every 12 hours~~ for each accumulator. ~~This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Main control board alarms are also available for these accumulator parameters.~~ The level transmitters for the accumulators measure the level over a 14" span for the corresponding 0-100% level indicated on the main control board. ~~Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.~~ INSERT 3

SR 3.5.1.3

See SR 3.5.1.2

SR 3.5.1.4

The boron concentration shall be verified to be within required limits for each accumulator ~~every 12 hours by monitoring inleakage~~. This is accomplished by monitoring the level in each accumulator ~~every 12 hours~~ and comparing to the previous level readings. An unexplained increase in level could be an indication of inleakage and, therefore, dilution of the boron concentration. If an unexplained increase in level is detected, the ongoing change in boron concentration shall be determined by calculation. If the calculation indicates that the boron concentration had decreased to within 100 ppm of the lower limit, the affected accumulator shall be sampled to confirm boron concentration. ~~In addition, the accumulators shall be sampled every 6 months to confirm that the boron concentration, inferred from inleakage monitoring, remains within limits. Six months is reasonable for verification by sampling to determine that each accumulator's boron concentration is within the required limits, because the static design of the accumulators limits the ways in which the concentration can be changed. This Frequency is adequate to identify changes that could occur from mechanisms, such as stratification or inleakage.~~ INSERT 3

SR 3.5.1.5

Verification ~~every 31 days~~ that power is removed from each accumulator isolation valve operator when the pressurizer pressure is > 1600 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, no accumulators would be available for injection if the LOCA were to occur in the cold leg containing the only OPERABLE accumulator. ~~Since power is removed under administrative control and valve position is verified every 12 hours, the 31 day Frequency will provide adequate assurance that power is removed.~~ INSERT 3

B.1 and B.2

If the inoperable train cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

If both trains of ECCS are inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be immediately entered. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Use of control board indication for valve position is an acceptable verification. Misalignment of these valves could render both ECCS trains inoperable. The listed valves are secured in position by removal of AC power or key locking the DC control power. These valves are operated under administrative controls such that any changes with respect to the position of the valve breakers or key locks is unlikely. The verification of the valve breakers and key locks is performed by SR 3.5.2.3. Mispositioning of these valves can disable the function of both ECCS trains and invalidate the accident analyses. ~~A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned valve is unlikely.~~

INSERT 3

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. ~~The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position in most cases,~~

~~would only affect a single train. This Frequency has been shown to be acceptable through operating experience.~~ INSERT 3

SR 3.5.2.3

Verification ~~every 31 days~~ that AC or DC power is removed, as appropriate, for each valve specified in SR 3.5.2.1 ensures that an active failure could not result in an undetected misposition of a valve which affects both trains of ECCS. If this were to occur, no ECCS injection or recirculation would be available. ~~Since power is removed under administrative control and valve position is verified every 12 hours, the 31 day Frequency will provide adequate assurance that power is removed.~~ INSERT 3

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. This type of testing may be accomplished by measuring the pump developed head at a single point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.~~

INSERT 3

SR 3.5.2.6

See SR 3.5.2.5

SR 3.5.2.7

Periodic inspections of the containment sump suction inlet to the RHR System ensure that it is unrestricted and stays in proper operating condition. ~~The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the need to have access to the location. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.~~ INSERT 3

REFERENCES

1. Letter from R. A. Purple, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment 7 to Provisional Operating License No. DPR-18," dated May 14, 1975.

2. Branch Technical Position (BTP) ICSB-18, "Application of the Single Failure Criterion to Manually-Controlled Electrically Operated Valves."

3. Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: "Issuance of Amendment No. 42 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant (TAC No. 79829)," dated June 3, 1991.

4. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic VI-7.B: ESF Switchover from Injection to Recirculation Mode, Automatic ECCS Realignment, Ginna," dated December 31, 1981.

5. NUREG-0821.

6. UFSAR, Section 6.3.

7. Not Used

8. Atomic Industrial Forum (AIF) GDC 44, Issued for comment July 10, 1967.

9. 10 CFR 50.46.

10. UFSAR, Section 15.6.

11. UFSAR, Section 6.2.

ACTIONS

A.1

With RWST boron concentration not within limits, it must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the CS System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST boron concentration to within limits was developed considering the time required to change the boron concentration and the fact that the contents of the tank are still available for injection.

B.1

With the RWST water volume not within limits, it must be restored to OPERABLE status within 1 hour. In this Condition, neither the ECCS nor the CS System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST water volume should be verified ~~every 7 days~~ to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and CS System pump operation on recirculation. ~~Since the RWST volume is normally stable and the RWST is located in the Auxiliary Building which provides sufficient leak detection capability, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.~~

INSERT 3

SR 3.5.4.2

The boron concentration of the RWST should be verified ~~every 7 days~~ to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. ~~Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.~~ INSERT 3

REFERENCES

1. UFSAR, Section 3.11.
2. 10 CFR 50.49.
3. UFSAR, Section 6.3 and Chapter 15.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. ~~Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when the containment airlock door is opened, this test is only required to be performed once every 24 months. The 24 month Frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.~~ INSERT 3

REFERENCES

1. UFSAR, Section 6.2.1.1.
2. 10 CFR 50, Appendix J, Option B.

E.2

Required Action E.2 requires that the mini-purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated within 1 hour. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure (including a single human error). For automatic valves, this requires two independent means to prevent the valve from re-opening. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A purge valve with resilient seals utilized to satisfy Required Action E.2 must have been demonstrated to meet the leakage requirements of SR 3.6.3.5. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a major violation of containment does not exist.

Following completion of Required Action E.1, verification that the affected penetration flow path remains isolated must be performed in accordance with Required Action D.2.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

This SR ensures that the mini-purge valves are closed except when the valves are opened under administrative control. The mini-purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, maintenance activities, operational requirements, or for Surveillances that require the valves to be open. To be opened, the valves must be capable of closing under accident conditions, the containment isolation signal to the valves must be OPERABLE, and the effluent release must be monitored to ensure that it remains within regulatory limits. ~~The 31 day Frequency is based on the relative importance of these valves since they provide a direct path to the outside environment and the administrative controls that are in place.~~

INSERT 3

SR 3.6.3.2

This SR requires verification that each containment isolation boundary located outside containment and not locked, sealed or otherwise secured in the required position is performing its containment isolation accident function. Containment isolation boundaries located beneath Appendix R fire wrap may be considered secured in the required position due to the administrative controls in place provided that a verification of the boundary position was made prior to securing the fire wrap. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment barrier is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation boundaries outside containment and capable of being mispositioned are in the correct position. This includes manual valves, blind flanges, pipe and end caps, and closed systems. ~~Since containment isolation boundaries are maintained under administrative controls with containment isolation boundary tags installed, the probability of their misalignment is low and a 92 day Frequency to verify their correct position is appropriate.~~ The SR specifies that isolation boundaries that are open under administrative controls are not required to meet the SR during the time the boundaries are open.

INSERT 3

The SR is modified by two notes. The first Note applies to containment isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means. Allowing verification by administrative means (e.g., procedure control) is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these isolation boundaries, once they have been verified to be in the proper position, is small. The Second Note states that this SR is not applicable to containment isolation boundaries which receive an automatic signal since the isolation times of these valves are verified by SR 3.6.3.4 and the boundaries are required to be OPERABLE.

SR 3.6.3.3

This SR requires verification that each containment isolation boundary located inside containment and not locked, sealed or otherwise secured in the required position is performing its containment isolation accident function. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment barrier is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation boundaries inside containment and capable of being mispositioned are in the correct position. This includes manual valves, blind flanges, pipe and end caps, and closed systems. Since containment isolation boundaries are maintained under administrative controls with containment isolation boundary tags installed, the

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

INSERT 3

REFERENCES

1. Atomic Industry Forum GDC 53 and 57, issued for comment July 10, 1967.
2. Branch Technical Position CSB 6-4, "Containment Purging During Normal Operation."
3. UFSAR, Section 6.2.4 and Table 6.2-15.
4. Regulatory Guide 1.4, Revision 2.
5. 10 CFR 50, Appendix A, GDC 55, 56, and 57.
6. Ginna Station Procedure A-3.3.
7. NUREG-0800, Section 6.2.4.

to OPERABLE status within 1 hour. However, due to the large containment free volume and limited size of the containment Mini-Purge System, 8 hours is allowed to restore containment pressure to within limits. This is justified by the low probability of a DBA during this time period.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that plant operation remains within the limits assumed in the containment analysis. This verification should normally be performed using PI-944. ~~The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.~~ INSERT 3

Calibration of PI-944 or other containment pressure monitoring devices should be performed in accordance with industry standards.

REFERENCES

1. UFSAR, Section 6.2.1.2.
2. 10 CFR 50, Appendix K.

containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within the limit within 24 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 24 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. There are 6 containment air temperature indicators (TE-6031, TE-6035, TE-6036, TE-6037, TE-6038, and TE-6045) such that a minimum of three should be used for calculating the arithmetic average. ~~The 12 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal containment temperature condition.~~ INSERT 3

Calibration of these temperature indicators shall be performed in accordance with industry standards.

D.1

With one or two CRFC units inoperable, the inoperable CRFC unit(s) must be restored to OPERABLE status within 7 days. The inoperable CRFC units provided up to 100% of the containment heat removal needs. The 7 day Completion Time is justified considering the redundant heat removal capabilities afforded by combinations of the CS System and CRFC System and the low probability of DBA occurring during this period.

E.1 and E.2

If the Required Action and associated Completion Time of Condition D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With two CS trains inoperable, or three or more CRFC units inoperable, the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

The applicable SR descriptions from Bases 3.5.2 apply. This SR is required since the OPERABILITY of valves 896A and 896B is also required for the CS System.

SR 3.6.6.2

Verifying the correct alignment for manual, power operated, and automatic valves in the CS flow path provides assurance that the proper flow paths will exist for CS System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (there are no valves inside containment) and capable of potentially being mispositioned are in the correct position.

INSERT 3

SR 3.6.6.3

Verifying the correct alignment for manual, power operated, and automatic valves in the NaOH System flow path provides assurance that the proper flow paths will exist for NaOH System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (there are no valves inside containment) and capable of potentially being mispositioned are in the correct position. INSERT 3

SR 3.6.6.4

Operating each CRFC unit for ≥ 15 minutes ~~once every 31 days~~ ensures that all CRFC units are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, damper failures, or excessive vibration can be detected for corrective action. The A and C CRFC units must be operated with their respective charcoal filter train in the post accident alignment. ~~The 31 day Frequency was developed considering the known reliability of the fan units and controls, the redundancy available, and the low probability of significant degradation of the CRFC units occurring between surveillances. It has also been shown to be acceptable through operating experience.~~ INSERT 3

SR 3.6.6.5

Verifying cooling water (i.e., SW) flow to each CRFC unit provides assurance that the energy removal capability of the CRFC assumed in the accident analyses will be achieved (Ref. 11). The minimum and maximum SW flows are not required to be specifically determined by this SR due to the potential for a containment air temperature transient. Instead, this SR verifies that SW flow is available to each CRFC unit. ~~The 31 day Frequency was developed considering the known reliability of the SW System, the two CRFC train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.~~ INSERT 3

SR 3.6.6.6

Verifying each CS pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by the ASME Code (Ref. 12). Since the CS pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice testing confirms component OPERABILITY, trends performance, and detects incipient failures by

abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.6.7

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water that is injected. This SR is performed to verify the availability of sufficient NaOH solution in the spray additive tank. ~~The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval since the tank is normally isolated.~~ INSERT 3 Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.6.8

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. ~~The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration since the tank is normally isolated and the probability that any substantial variance in tank volume will be detected.~~ INSERT 3

SR 3.6.6.9

This SR verifies that the required CRFC unit testing is performed in accordance with the VFTP. The VFTP includes testing HEPA filter performance. The minimum required flow rate through each of the four CRFC units is 33,000 cubic feet per minute at accident conditions (or 38,500 cubic feet per minute at normal operating conditions). Specific test frequencies and additional information are discussed in detail in the VFTP. However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles as defined by Regulatory Guide 1.52 (Ref. 13).

SR 3.6.6.10

These SRs require verification that each automatic CS valve in the flowpath (860A and 860D) actuates to its correct position and that each CS pump starts upon receipt of an actual or simulated actuation of a containment High pressure signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the~~

~~Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ INSERT 3

SR 3.6.6.11

See SR 3.6.6.10

SR 3.6.6.12

This SR requires verification that each CRFC unit, and the charcoal filter train associated with the A and C units, actuates upon receipt of an actual or simulated safety injection signal. ~~The 24 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.10 and SR 3.6.6.11, above, for further discussion of the basis for the 24 month Frequency.~~ INSERT 3

SR 3.6.6.13

This SR provides verification that each automatic valve in the NaOH System flow path that is not locked, sealed, or otherwise secured in position (836A and 836B) actuates to its correct position upon receipt of an actual or simulated actuation of a containment Hi-Hi pressure signal. ~~The 24 month frequency is based on engineering judgement and has been shown to be acceptable through operating experience. See SR 3.6.6.10 and SR 3.6.6.11, above, for further discussion of the basis for the 24 month Frequency.~~ INSERT 3

SR 3.6.6.14

To ensure that the correct pH level is established in the borated water solution provided by the CS System, flow through the eductor is verified INSERT 1 ~~once every 5 years~~. This SR in conjunction with SR 3.6.6.13 provides assurance that NaOH will be added into the flow path upon CS initiation. A minimum flow of 20 gpm through the eductor must be established as assumed in the accident analyses. A flow path must also be verified from the NaOH tank to the eductors. ~~Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow injection.~~ INSERT 3

SR 3.6.6.15

With the CS inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. As an alternative, a visual inspection (e.g. boroscope) of the nozzles or piping could be utilized in lieu of an air or smoke test if a visual inspection is determined to provide an equivalent or a 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. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive

SR 3.7.2.3

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The MSIVs should not be tested at power, since even a partial stroke exercise increases the risk of a valve closure and plant transient when the plant is above MODE 4. As the MSIVs are not tested at power, they are exempt from the ASME Code (Ref. 5), requirements during operation in MODES 1, 2 and 3.

~~The frequency of MSIV testing is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.~~ INSERT 3

REFERENCES

1. UFSAR, Section 5.4.4.
2. UFSAR, Section 15.1.5.
3. UFSAR, Section 3.6.2.5.1.
4. 10 CFR 50.67.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a cooldown of the RCS, the ARVs must be able to be opened either remotely or locally. This SR ensures that the ARVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ARV during a plant cooldown may satisfy this requirement. ~~Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. The Frequency is acceptable from a reliability standpoint.~~ INSERT 3

SR 3.7.4.2

The function of the block valve is to isolate a failed open ARV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during plant cooldown may satisfy this requirement. ~~Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. The Frequency is acceptable from a reliability standpoint.~~ INSERT 3

REFERENCES

1. UFSAR, Section 10.3.2.5.

2. UFSAR, Section 15.6.3.

3. UFSAR, Section 15.1.6.

plant should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one MDAFW, TDAFW, or SAFW train to OPERABLE status. For the purposes of this Required Action, only one TDAFW train flow path and the pump must be restored to exit this Condition.

Required Action H.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one MDAFW, TDAFW, or SAFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW and SAFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.

~~The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~ INSERT 3

SR 3.7.5.2

Periodically comparing the reference differential pressure and flow of each AFW pump in accordance with the inservice testing requirements of the ASME Code (Ref. 4) detects trends that might be indicative of an incipient failure. The Frequency of this surveillance is specified in the Inservice Testing Program, which encompasses the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy this requirement.

This SR is modified by a Note indicating that the SR is only required to be met prior to entering MODE 1 for the TDAFW pump since suitable test conditions have not been established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.3

Periodically comparing the reference differential pressure and flow of each SAFW pump in accordance with the inservice testing requirements of the ASME Code (Ref. 4) detects trends that might be indicative of an incipient failure. Because it is undesirable to introduce SW into the SGs while they are operating, this testing is performed using the test condensate tank. The Frequency of this surveillance is specified in the Inservice Testing Program, which encompasses the ASME Code. The ASME Code provides the activities and Frequencies necessary to satisfy this requirement.

SR 3.7.5.4

This SR verifies that each AFW and SAFW motor operated suction valve from the SW System (4013, 4027, 4028, 9629A, and 9629B), each AFW and SAFW discharge motor operated valve (4007, 4008, 9701A, 9701B, 9704A, 9704B, and 9746), and each SAFW cross-tie motor operated valve (9703A and 9703B) can be operated when required. The Frequency of this Surveillance is specified in the Inservice Test Program and is consistent with the ASME Code (Ref. 4). The TDAFW discharge motor operated valve (3996) is maintained open and not required to be closed for the DBA's and transients described within the Applicable Safety Analyses section. Therefore, testing of the TDAFW discharge motor operating valve is not required.

SR 3.7.5.5

This SR verifies that AFW can be delivered to the appropriate SG in the event of any accident or transient that generates an actuation signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.~~ INSERT 3

SR 3.7.5.6

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an actuation signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. ~~The 24 month Frequency is based on the potential need to perform this Surveillance under the conditions that apply during a plant outage.~~ INSERT 3

This SR is modified by a Note indicating that the SR is only required to be met prior to entering MODE 1 for the TDAFW pump since suitable test conditions may have not been established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.7

This SR verifies that the SAFW System can be actuated and controlled from the control room. The SAFW System is assumed to be manually initiated within 14.5 minutes in the event that the preferred AFW System is inoperable. This Surveillance includes the verification of the automatic response of the motor operated discharge valves (9701A and 9701B) and the recirculation valves (9710A and 9710B). ~~The Frequency of 24 months is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed at power.~~

INSERT 3

REFERENCES

1. UFSAR, Section 10.5.
2. UFSAR Chapter 15.
3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

ACTIONS

A.1 and A.2

If the CST water volume is not within limits, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the preferred AFW pumps are OPERABLE and immediately available upon AFW initiation, and that the backup supply has the required volume of water available. Alternate sources of water include, but is not limited to, the SW System and the all-volatile-treatment condensate tank. In addition, the CSTs must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. Continued verification of the backup supply is not required due to the large volume of water typically available from these alternate sources. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CSTs.

B.1 and B.2

If the backup supply cannot be verified or the CSTs cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CSTs contain the required volume of cooling water. The 24,350 gal minimum volume is met if one CST is ≥ 22.8 ft (including instrument uncertainty) or if both CSTs are ≥ 13.6 ft (including instrument uncertainty). ~~The 12 hour Frequency is based on operating experience and the need for operator awareness of plant evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.~~ INSERT 3

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual and power operated valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.

~~The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~ INSERT 3

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCW loop header.

SR 3.7.7.2

This SR verifies that the two motor operated isolation valves to the RHR heat exchangers (738A and 738B) can be operated when required since the valves are normally maintained closed. The Frequency of this Surveillance is specified in the Inservice Test Program and is consistent with the ASME Code (Ref. 2).

REFERENCES

1. UFSAR, Section 9.2.2.
2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

C.1 and C.2

If the SW pumps cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

With three or more SW pumps or the loop header inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

Required Action D.1 is modified by a Note requiring that the applicable Conditions and Required Actions of LCO 3.7.7, "CCW System," be entered for the component cooling water heat exchanger(s) made inoperable by SW. This note is provided since the inoperable SW system may prevent the plant from reaching MODE 5 as required by LCO 3.0.3 if both CCW heat exchangers are rendered inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that adequate NPSH is available to operate the SW pumps and that the SW suction source temperature is within the limits assumed by the accident analyses and the system design. ~~The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.~~ INSERT 3

SR 3.7.8.2

Verifying the correct alignment for manual, power operated, and automatic valves in the SW flow path provides assurance that the proper flow paths exist for SW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.

~~The 31-day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.~~ INSERT 3

This SR is modified by a Note indicating that the isolation of the SW flow to individual components or systems may render those components inoperable, but does not affect the OPERABILITY of the SW System.

SR 3.7.8.3

This SR verifies that all SW loop header cross-tie valves are locked in the correct position. This includes verification that manual valves 4623, 4639, 4640, 4665, 4668B, 4669, 4756, and 4760 are locked open and that manual valves 4610, 4611, 4612, and 4779 are locked closed. The diesel generator cross-tie valves (4665, 4760, 4669, and 4668B) may be individually (one at a time) closed intermittently under administrative controls, such as during surveillance testing, as described in the LCO Bases. ~~The 31-day Frequency is based on engineering judgement, is consistent with the procedural controls governing locked valves, and ensures correct valve positions.~~ INSERT 3

SR 3.7.8.4

This SR verifies proper automatic operation of the SW motor operated isolation valves on an actual or simulated actuation signal (i.e., coincident safety injection and undervoltage signal). SW is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ INSERT 3

SR 3.7.8.5

This SR verifies proper automatic operation of the SW pumps on an actual or simulated actuation signal. This includes the actuation of the SW pumps following an undervoltage signal and following a coincident safety injection and undervoltage signal. SW is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. ~~The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~ INSERT 3

a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each CREATS filtration train ~~once every 31 days~~ for ≥ 15 minutes provides an adequate check of this system. ~~The 31 day Frequency is based on the reliability of the equipment, and the two train redundancy.~~ INSERT 3

SR 3.7.9.2

This SR verifies that the required CREATS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, flow rate, and the physical properties of the activated charcoal. The required flowrate through each CREATS filtration train is 6000 cubic feet per minute (±10%). Specific test Frequencies and additional information are discussed in detail in the VFTP.

The value of 1.5% methyl iodide penetration was chosen for the laboratory test sample acceptance criteria because, even though the new system contains 4-inch charcoal beds, the design face velocity is 61 fpm. Regulatory Guide 1.52, Revision 3 (Ref. 9), Table 1, provides testing criteria assuming a 40 fpm face velocity. The value of 1.5% was interpolated between the two values listed because of the higher face velocity of Ginna's system. The face velocity is listed in the specification because it is a non standard number. Testing at 61 fpm or greater satifies the criteria.

SR 3.7.9.3

This SR verifies that each CREATS train starts and operates and that each CREATS automatic damper actuates on an actual or simulated actuation signal. ~~The Frequency of 24 months is based on Industry operating experience.~~ INSERT 3

SR 3.7.9.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA

ACTIONS

A.1

When the ABVS is inoperable, action must be taken to place the plant in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the Auxiliary Building. This does not preclude the movement of fuel to a safe position.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies in the Auxiliary Building which have decayed < 60 days since being irradiated, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

This SR verifies the OPERABILITY of the ABVS. During fuel movement operations, the ABVS is designed to maintain a slight negative pressure in the Auxiliary Building to prevent unfiltered LEAKAGE. This SR ensures that Auxiliary Building exhaust fan C, and either Auxiliary Building main exhaust fan A or B are in operation and that the ABVS interlock mode switch is in the correct position. ~~The Frequency of 24 hours is based on engineering judgement and shown to be acceptable through operating experience.~~ INSERT 3

SR 3.7.10.2

This SR verifies the integrity of the Auxiliary Building enclosure. The ability of the Auxiliary Building to maintain negative pressure with respect to the uncontaminated outside environment must be periodically verified to ensure proper functioning of the ABVS. During fuel movement operations, the ABVS is designed to maintain a slight negative pressure in the Auxiliary Building to prevent unfiltered leakage. This SR ensures that a negative pressure is being maintained in the Auxiliary Building. ~~The Frequency of 24 hours is based on engineering judgement and shown to be acceptable through operating experience.~~ INSERT 3

SR 3.7.10.3

This SR verifies that the required SFP Charcoal Adsorber System testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SFP Charcoal Adsorber System filter tests are in general accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes

ACTIONS

A.1

When the initial conditions assumed in the fuel handling accident analysis cannot be met, steps should be taken to preclude the accident from occurring. When the SFP water level is lower than the required level, the movement of irradiated fuel assemblies in the SFP is immediately suspended. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position (e.g., movement to an available rack position).

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since if moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies sufficient SFP water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically during movement of irradiated fuel assemblies to ensure the fuel handling accident assumptions are met. ~~The 7 day Frequency is appropriate because the volume in the pool is normally stable and the SFP is designed to prevent drainage below 23 ft. Water level changes are controlled by plant procedures and are acceptable based on operating experience.~~ INSERT 3

Verification of SFP water level can be accomplished by several means. The top of the upper SFP pump suction line is 23 ft above the fuel stored in the pool. If there is ≥ 23 ft of water above the reactor vessel flange (as required by LCO 3.9.6), with equal pressure in the containment and the Auxiliary Building, then at least 23 ft of water is available above the top of the active fuel in the storage racks.

In addition to the physical design features, there are two SFP level alarms (LAL 634) which are available to alert the operators of changing SFP level. A low level alarm will actuate when the SFP water level falls 4 inches or more from the normal level while a high level alarm will actuate when the SFP water level rises 4 inches or more from the normal level. These alarms must receive a calibration consistent with industry practices before they are to be used to meet this SR.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the SFP to ensure the SFP k_{eff} remains ≤ 0.95 at all times.

ACTIONS

A.1 and A.2

When the concentration of boron in the SFP is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The initiation of actions to restore concentration of boron is simultaneous with suspending movement of fuel assemblies.

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since if the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that the concentration of boron in the SFP is within the limit. As long as this SR is met, the analyzed accidents are fully addressed. ~~The 7 day Frequency is appropriate since the boron is credited with maintaining the SFP subcritical. Also, the volume and boron concentration in the pool is normally stable and all water level changes and boron concentration changes are controlled by plant procedures.~~ INSERT 3

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity is not within limits the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. ~~The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.~~

INSERT 3

REFERENCES

1. 10 CFR 50.67.
2. Design Analysis DA-NS-2002-007, Main Steam Line Break Offsite and Control Room Doses.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function (Ref. 2). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions).

SR 3.8.1.1

This SR ensures proper circuit continuity for the independent offsite power source to each of the onsite 480 V safeguards buses and availability of offsite AC electrical power. Checking breaker alignment and indicated power availability verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their qualified power source. ~~The Frequency of 7 days is adequate since breaker position is not likely to change without the operators knowledge and because alarms and indications of breaker status are available in the control room.~~ INSERT 3

SR 3.8.1.2

This SR verifies that each DG starts from standby conditions and achieves rated voltage and frequency. This ensures the availability of the DG to mitigate DBAs and transients and to maintain the plant in a safe shutdown condition. The DG voltage control may be either in manual or automatic during the performance of this SR. ~~The Frequency of 31 days is adequate to provide assurance of DG OPERABILITY, while minimizing degradation resulting from testing.~~ INSERT 3

This SR is modified by two Notes. Note 1 indicates that performance of SR 3.8.1.9 satisfies this SR since SR 3.8.1.9 is a complete test of the DG. The second Note states that all DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. This minimizes the wear on moving parts that do not get lubricated when the engine is not running.

SR 3.8.1.3

This SR verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures. A maximum run time of < 120 minutes minimizes the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.85 lagging and 0.95 lagging. The upper load band limit of < 2250 kW is the DG two-hour rating and is provided to avoid routine overloading of the DG which may result in more frequent inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The lower band limit of 2025 kW bounds the maximum expected load following a DBA, based on worst case loading during the injection phase of the accident. The diesel generator loading will be below the long-term rating of 1950 kW within two hours.

In addition to verifying the DG capability for synchronizing with the offsite electrical system and accepting loads, the DG ventilation system should also be verified during this surveillance.

INSERT 3

~~The Frequency of 31 days is adequate to provide assurance of DG OPERABILITY, while minimizing degradation resulting from testing.~~

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients outside the load band (e.g., due to changing bus loads), do not invalidate this test. Similarly, momentary power factor transients above or below the administrative limit do not invalidate the test. Note 3 indicates that this Surveillance shall be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful performance of SR 3.8.1.2 or SR 3.8.1.9 must precede this surveillance to prevent unnecessary starts of the DGs.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in each day tank is at or above the minimum level, including instrument uncertainty, at which fuel oil is automatically added when the fuel oil transfer pump is in auto and the DG is operating. This level ensures adequate fuel oil for a minimum of 1 hour of DG operation at 110% of full load. A level of 8.75 inches, as read on the local sight glass, achieves these requirements.

INSERT 3

~~The Frequency of 31 days is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and operators would be aware of any large uses of fuel oil during this period.~~

SR 3.8.1.5

This SR demonstrates that each DG fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of the DGs. This

Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic or manual fuel transfer systems are OPERABLE.

~~The Frequency of 31 days is adequate to provide assurance of DG OPERABILITY, since the design of the fuel oil transfer system is such that pumps operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day tanks during or following DG operation.~~ INSERT 3

SR 3.8.1.6

This SR involves the transfer of the 480 V safeguards bus power supply from the 50/50 mode to the 100/0 mode and 0/100 mode which demonstrates the OPERABILITY of the alternate circuit distribution network to power the required loads. ~~The Frequency of 24 months is based on engineering judgment, taking into consideration the plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ INSERT 3

SR 3.8.1.7

This SR verifies that each DG does not trip during and following a load rejection of ≥ 295 kW. Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This SR demonstrates the DG load response characteristics and capability to reject the largest single load on the buses supplied by the DG (i.e., a safety injection pump).

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing must be performed using a power factor ≤ 0.9 lagging. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

~~The Frequency of 24 months is based on engineering judgement, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~ INSERT 3

This SR is modified by two Notes. The first Note states that this Surveillance shall not be performed in MODE 1, 2, 3, or 4. The reason for the Note is that during operation in these MODES, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant

safety systems. The second Note acknowledges that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.8

This SR demonstrates that DG noncritical protective functions (e.g., overcurrent, reverse power, local stop pushbutton) are bypassed on an actual or simulated SI actuation signal. The noncritical trips are bypassed during DBAs but still provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG. The DG critical protective functions (engine overspeed, low lube oil pressure, and start failure (overcrank) relay) will be tested periodically per the station periodic maintenance program.

~~The Frequency of 24 months is based on engineering judgment, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.~~ INSERT 3

This SR is modified by two Notes. The first Note states that this Surveillance shall not be performed in MODE 1, 2, 3, or 4. The reason for the Note is that performing the Surveillance would remove a required DG from service which is undesirable in these MODES. The second Note acknowledges that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.9

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This SR demonstrates the DG operation during an actual or simulated loss of offsite power signal in conjunction with an actual or simulated SI actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

Since it is not possible to operate all sequenced motors at their DBA loadings, a transient simulation program is used to demonstrate acceptable DG governor and voltage regulator operation. To successfully validate the testing data with the transient simulation program, the largest loads (with respect to both kW and current) must be sequenced on the

DG during performance of this test. This includes two SI pumps, a CS and RHR pump, and safety-related motor control centers, as a minimum.

INSERT 3

~~The Frequency of 24 months is based on engineering judgement, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.~~

This SR is modified by three Notes. Note 1 states that all DG starts may be preceded by an engine prelube period which is intended to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine lube oil continuously circulated and temperature maintained consistent with manufacturer recommendations for the DGs. Note 2 states that this Surveillance shall not be performed in MODE 1, 2, 3, or 4 since performing the Surveillance during these MODES would remove a required offsite circuit from service, cause perturbations to the electrical distribution systems, and challenge safety systems. Note 3 acknowledges that credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. UFSAR, Chapter 8.
2. Atomic Industrial Forum (AIF) GDC 39, Issued for comment July 10, 1967.
3. UFSAR, Section 9.4.9.5.
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 15.
6. 10 CFR 50, Appendix A, GDC 17.
7. "American National Standard, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
8. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
9. UFSAR Section 3.11

time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample practices (bottom sampling), contaminated sampling equipment, or errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling and re-analysis of the DG fuel oil.

C.1

With the new fuel oil properties defined in SR 3.8.3.2 not within required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

D.1

With a Required Action and associated Completion Time not met, or one or more DG's fuel oil notwithin limits for reasons other than addressed by Conditions A, B, or C (e.g., cloud point temperature reached), the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR verifies an onsite supply of ≥ 5000 gal of fuel oil is available for each required DG. This ensures that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 40 hours while providing maximum post-LOCA loads. The 40 hour period is sufficient time to place the plant in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

~~The Frequency of 31 days is adequate to ensure that a sufficient supply of fuel oil is available, since indications are available to ensure that operators would be aware of any large uses of fuel oil during this period.~~

INSERT 3

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The elevated equalize charge capability of the battery chargers is not an OPERABILITY requirement of the battery chargers and is not to be in service during the surveillance. The voltage drop when changing from the equalize conditions to the normal float conditions occurs relatively quickly. ~~The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 8).~~ INSERT 3

SR 3.8.4.2

This SR verifies that the capacity of each battery is adequate to supply and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test. A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length corresponds to the design duty cycle requirements specified in Reference 2.

~~The Surveillance Frequency of 24 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 9) and Regulatory Guide 1.129 (Ref. 10), which state that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests not to exceed 24 months.~~ INSERT 3

This SR is modified by two Notes. Note 1 states that SR 3.8.4.3 may be performed in lieu of SR 3.8.4.2. This substitution is acceptable because SR 3.8.4.3 represents a more severe test of battery capacity than does SR 3.8.4.2. Note 2 states that this surveillance shall not be performed in MODE 1, 2, 3, or 4 because performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

SR 3.8.4.3

This Surveillance verifies that each battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test. A battery performance test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity as determined by specified acceptance

criteria. The test is intended to determine overall battery degradation due to age and usage.

A battery should be replaced if its capacity is below 80% of the manufacturer rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

INSERT 1

The Frequency for this SR is ~~60 months~~ when the battery is < 85% of its expected life with no degradation and 12 months if the battery shows degradation or has reached 85% of its expected life with a capacity < 100% of the manufacturer's rating. When the battery has reached 85% of its expected life with capacity ≥ 100% of the manufacturer's rating, the Frequency becomes 24 months. Battery degradation is indicated when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer rating. These Frequencies are considered acceptable based on the testing being performed in a conservative manner relative to the battery life and degradation. This ensures that battery capacity is adequately monitored and that the battery remains capable of performing its intended function. INSERT 3

This SR is modified by a Note stating that this SR shall not be performed in MODE 1, 2, 3, or 4. The reason for the Note is that during operation in these MODES, performance of this SR could cause perturbations to the electrical distribution system and challenge safety systems.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 39, Issued for comment July 10, 1967.
2. UFSAR, Section 8.3.2.
3. UFSAR, Section 9.4.9.3.
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 15.
6. UFSAR, Section 8.3.1.
7. 10 CFR 50, Appendix A, GDC 17.
8. IEEE-450-1980.
9. ~~Regulatory Guide 1.32, February 1977.~~ Deleted
10. ~~Regulatory Guide 1.129, December 1974.~~ Deleted

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that the electrolyte level of each connected battery cell is above the top of the plates and not overflowing. This is consistent with IEEE-450 (Ref. 4) and ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. ~~The Frequency of 31 days is consistent with IEEE-450.~~ INSERT 3

SR 3.8.6.2

This SR verifies that the float voltage of each connected battery cell is > 2.07 V. This limit is based on IEEE-450 (Ref. 4) which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement. ~~The frequency of 31 days is also consistent with IEEE-450.~~ INSERT 3

SR 3.8.6.3

This SR verifies the specific gravity of the designated pilot cell in each battery is ≥ 1.195. This value is based on manufacturer recommendations. According to IEEE-450 (Ref. 4), the specific gravity readings are based on a temperature of 77°F (25°C). The specific gravity readings are corrected for actual electrolyte temperature. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is further discussed in IEEE-450. ~~The Frequency of 31 days is consistent with IEEE-450.~~ INSERT 3

SR 3.8.6.4

This SR verifies the average electrolyte temperature of the designated pilot cell in each battery is ≥ 55°F. This temperature limit is an initial assumption of the battery capacity calculations. ~~The Frequency of 31 days is consistent with IEEE-450 (Ref. 4).~~ INSERT 3

SR 3.8.6.5

This SR verifies that the average temperature of every fifth cell of each battery is ≥ 55°F. This is consistent with the recommendations of IEEE-450 (Ref. 4). Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. ~~The Frequency of 92 days is consistent with IEEE-450.~~ INSERT 3

SR 3.8.6.6

This SR verifies the specific gravity of each connected cell is not more than 0.020 below average of all connected cells and that the average of all connected cells is ≥ 1.195. This value is based on manufacturer recommendations and IEEE-450 (Ref. 4) which ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery. The temperature correction for specific gravity readings is the same as that discussed for SR 3.8.6.3. ~~The Frequency of 92 days is consistent with IEEE-450.~~ INSERT 3

REFERENCES

1. UFSAR, Section 3.8.2.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.
4. IEEE-450-1980.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This SR verifies correct static switch alignment to Instrument Bus A and C. This verifies that the inverters are functioning properly and AC Instrument Bus A and C are energized from their respective inverter. The verification ensures that the required power is available for the instrumentation of the RPS and ESF connected to the AC instrument buses. ~~The Frequency of 7 days takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.~~ INSERT 3

SR 3.8.7.2

This SR verifies the correct Class 1E CVT alignment to Instrument Bus B. This verifies that the Class 1E CVT is functioning properly and supplying power to AC Instrument Bus B. The verification ensures that the required power is available for the instrumentation of the RPS and ESF connected to the AC instrument bus. ~~The Frequency of 7 days takes into account the redundant instrument buses and other indications available in the control room that alert the operator to the Class 1E CVT malfunctions.~~ INSERT 3

REFERENCES

1. UFSAR, Chapter 8.3.2.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.
4. UFSAR, Section 8.3.1.
5. 10 CFR 50, Appendix A, GDC 17.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC instrument bus power source should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power or powered from an alternate power source.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This SR verifies correct static switch alignment to the required AC instrument buses. This SR verifies that the inverter is functioning properly and the AC instrument bus is energized from the inverter. The verification ensures that the required power is available for the instrumentation connected to the AC instrument bus. ~~The Frequency of 7 days takes into account the redundant capability of the inverter and other indications available in the control room that alert the operator to inverter malfunctions.~~ INSERT 3

SR 3.8.8.2

This SR verifies the correct Class 1E CVT alignment when Instrument Bus B is required. This verifies that the Class 1E CVT is functioning properly and supplying power to AC Instrument Bus B. The verification ensures that the required power is available for the instrumentation of the RPS and ESF connected to the AC instrument bus. ~~The Frequency of 7 days takes into account the redundant instrument buses and other indications available in the control room that alert the operator to the Class 1E CVT malfunctions.~~ INSERT 3

REFERENCES

1. None.

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

With two trains with inoperable electrical power distribution subsystems, the potential for a loss of safety function is greater. If a loss of safety function exists, no additional time is justified for continued operation and LCO 3.0.3 must be entered. This Condition may be entered with the loss of two trains of the same electrical power distribution subsystem, or with loss of Train A of one electrical power distribution subsystem coincident with the loss of Train B of a second electrical power distribution subsystem such that a loss of safety function exists.

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This SR verifies that the electrical power trains are functioning properly, with all required power source circuit breakers closed, tie-breakers open, and the buses energized from their allowable power sources. Required voltage for the AC electrical power distribution subsystem is ≥ 420 VAC; for the DC electrical power distribution subsystem, ≥ 108.6 VDC and ≤ 140 VDC; and for AC instrument bus electrical power distribution subsystem, between 113 VAC and 123 VAC at the instrument buses. Required voltage for the instrument distribution panels is between 110 VAC and 123 VAC. Required voltage for inverter MQ-483 is between 107 volts and 129.8 volts. The loss of inverter MQ-483 is addressed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation" and LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation" for the affected individual containment wide range pressure and steam generator B pressure instrumentation (PT-950 and PT-479). The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. ~~The Frequency of 7 days takes into account the redundant capability of the AC, DC, and AC instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.~~

INSERT 3

Therefore, Required Action A.2.5 requires declaring RHR inoperable, which results in taking the appropriate RHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the electrical power distribution trains are functioning properly, with all the required power source circuit breakers closed, required tie-breakers open, and the required buses energized from their allowable power sources. Required voltage for the AC power distribution electrical subsystem is ≥ 420 VAC, for the DC power distribution electrical subsystem ≥ 108.6 VDC and ≤ 140 VDC, and for AC instrument bus power distribution electrical subsystem is between 113 VAC and 123 VAC at the instrument buses. Required voltage for the instrument distribution panels is between 110 VAC and 123 VAC. Required voltage for inverter MQ-483 is between 107 volts and 129.8 volts. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. ~~The Frequency of 7 days takes into account the capability of the AC, DC, and AC instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.~~ INSERT 3

REFERENCES

1. None.

There are no safety analysis assumptions of boration flow rate and concentration that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for plant conditions.

Once action has been initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration of the refueling canal, the refueling cavity, and the portions of the RCS that are hydraulically coupled, is within the COLR limits. The boron concentration of the coolant is determined by chemical analysis. The sample should be representative of the portions of the RCS, the refueling canal, and the refueling cavity that are hydraulically coupled with the reactor core.

~~A Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of the representative sample(s). The Frequency is based on operating experience, which has shown 72 hours to be adequate.~~

INSERT 3

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27, Issued for comment July 10, 1967.

2. UFSAR, Section 15.4.4.2.

3. NUREG-0800, Section 15.4.6.

The Completion Time of 4 hours is sufficient to obtain and analyze coolant samples for boron concentration. The Frequency of once per 12 hours ensures unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

C.1, C.2, and C.3

With no audible count rate available, only visual indication is available and prompt and definite indication of a boron dilution event has been lost. Therefore, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Actions C.1 and C.2 shall not preclude completion of movement of a component to a safe position (i.e., other than a normal cooldown of the coolant volume for the purpose of system temperature control within established procedures).

Since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the audible count rate capability is restored. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion time of 4 hours is sufficient to obtain and analyze coolant samples for boron concentration. The Frequency of once per 12 hours ensures unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

This SR is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one monitor to a similar parameter on another monitor. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range monitors, but each monitor should be consistent with its local conditions.

The inoperability of one source range neutron flux channel prevents performance of a CHANNEL CHECK for the operable channel. However, the Required Actions for the inoperable channel requires suspension of CORE ALTERATIONS and positive reactivity addition such that the CHANNEL CHECK of the operable channel can consist of ensuring consistency with known core conditions.

~~The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."~~ INSERT 3

SR 3.9.2.2

This SR is the performance of a CHANNEL CALIBRATION ~~every 24 months~~. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to baseline data. ~~The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.~~ INSERT 3

REFERENCES

1. UFSAR, Section 7.7.3.2.
2. Atomic Industrial Forum (AIF) GDC 13 and 19, Issued for Comment July 10, 1967.

ACTIONS

A.1 and A.2

If the containment equipment hatch (or its closure plate or roll up door and associated enclosure building), air lock doors, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Ventilation Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the plant must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This SR demonstrates that each of the containment penetrations are in the required status. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked or otherwise prevented from closing (e.g., solenoid unable to vent).

~~The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.~~

INSERT 3

SR 3.9.3.2

This SR demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. ~~The 24 month Frequency maintains consistency with other similar instrumentation and valve testing requirements. In LCO 3.3.5, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 24 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months an ACTUATION LOGIC TEST and CHANNEL CALIBRATION is performed. These Surveillances will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.~~

INSERT 3

A.4

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

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

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This SR requires verification ~~every 12 hours~~ that one RHR loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal capability and mixing of the borated coolant to prevent thermal and boron stratification in the core. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.~~ INSERT 3

REFERENCES

1. UFSAR, Section 5.4.5.
2. UFSAR, Section 15.4.4.2.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This SR requires verification ~~every 12 hours~~ that one RHR loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal capability and mixing of the borated coolant to prevent thermal and boron stratification in the core. ~~The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.~~ INSERT 3

SR 3.9.5.2

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the standby pump. ~~The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.~~ INSERT 3

REFERENCES

1. UFSAR, Section 5.4.5.

2. UFSAR, Section 15.4.4.

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits and preserves the assumptions of the fuel handling accident analysis (Ref. 1). As such, it is the minimum required level during movement of fuel assemblies within containment. Maintaining this minimum water level in the refueling cavity also ensures that ≥ 23 ft of water is available in the spent fuel pool during fuel movement assuming that containment and Auxiliary Building atmospheric pressures are equal.

APPLICABILITY

This LCO is applicable when moving irradiated fuel assemblies within containment. This LCO is also applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts. The LCO ensures a sufficient level of water is present in therefueling cavity to minimize the radiological consequences of a fuel handling accident in containment. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.11, "Spent Fuel Pool (SFP) Water Level."

ACTIONS

A.1 and A.2

When the initial condition assumed in the fuel handling accident cannot be met, steps should be taken to preclude the accident from occurring. With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum refueling cavity water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 1).

~~The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.~~

INSERT 3

REFERENCES

1. UFSAR, Section 15.7.3.
2. 10 CFR 50.67.
3. Regulatory Guide 1.183.

ATTACHMENT 5

License Amendment Request

R. E. Ginna Nuclear Power Plant
Docket No. 50-244

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-1431) vs. Ginna Cross-Reference

TSTF-425 (NUREG-1431) vs. Ginna Cross-Reference

Technical Specification Section Title/Surveillance Description*	TSTF-425	GINNA
Definitions	**1.1**	**1.1**
Staggered Test Testing		
Shutdown margin (SDM)	**3.1.1**	**3.1.1**
Verify SDM within limits	3.1.1.1	3.1.1.1
Core Reactivity	**3.1.2**	**3.1.2**
Verify core reactivity within predicted value	3.1.2.1	3.1.2.2
Rod Group Alignment Limits	**3.1.4**	**3.1.4**
Verify individual rod position within alignment	3.1.4.1	3.1.4.1
Verify rod freedom of movement (trippability)	3.1.4.2	3.1.4.3
Verify individual rod position within alignment with rod position monitor inoperable	-------------	3.1.4.2
Shutdown Bank Insertion Limits	**3.1.5**	**3.1.5**
Verify shutdown bank within insertion limit per COLR	3.1.5.1	3.1.5.1
Control Bank Insertion Limit	**3.1.6**	**3.1.6**
Verify control bank within insertion limit per COLR	3.1.6.2	3.1.6.2
Verify sequence and overlap limits per COLR	3.1.6.3	3.1.6.4
Verify control bank within insertion limit per COLR when insertion limit monitor inoperable	-------------	3.1.6.3
Physics Tests (Exceptions Mode 2)	**3.1.8**	**3.1.8**
Verify RCS lowest loop temperature	3.1.8.2	3.1.8.2
Verify thermal power	3.1.8.3	3.1.8.3
Verify SDM within COLR	3.1.8.4	3.1.8.4
$F_q(Z)$	**3.2.1**	-------------
Verify measured $F_q(Z)$	3.2.1.1	-------------
Verify measured $F^W_{xy} < F^l_{xy}$	3.2.1.2	-------------
$F_q(Z)$ (RAOC-W(Z) Methodology	**3.2.1B**	**3.2.1**
Verify $F^c_q(Z)$ is within limit	3.2.1.1	3.2.1.1
Verify $F^W_q(Z)$ is within limit	3.2.1.2	3.2.1.2
$F_q(Z)$ (CAOC-W(Z) Methodology	**3.2.1C**	-------------
Verify $F^c_q(Z)$ is within limit	3.2.1.1	-------------
Verify $F^W_q(Z)$ is within limit	3.2.1.2	-------------
Nuclear Enthalpy Rise Hot Channel Factor (F^N_{dh})	**3.2.2**	**3.2.2**
Verify F_{dh} within limits per COLR	3.2.2.1	3.2.2.1
Verify F_{dh} within limits per COLR (only when one power range inoperable)	-------------	3.2.2.2
AFD	**3.2.3A/B**	-------------
Verify AFD is within limits	3.2.3.1	-------------
Update target flux difference	3.2.3.2	-------------
Determine by measurement target flux difference	3.2.3.3	-------------
AFD(RAOC Methodology)	**3.2.3B**	**3.2.3**
Verify AFD is within limits	3.2.3.1	3.2.3.1
QPTR	**3.2.4**	**3.2.4**
Verify QPTR within limits by calculation	3.2.4.1	3.2.4.1
Verify QPTR within limits by incore detectors	3.2.4.2	3.2.4.2

Technical Specification Section Title/Surveillance Description*	TSTF-425	GINNA
RTS Instrumentation	**3.3.1**	**3.3.1**
Channel Check	3.3.1.1	3.3.1.1
Compare calorimetric to power range channel output	3.3.1.2	3.3.1.2
Compare results of incore to NIS	3.3.1.3	3.3.1.3
Perform TADOT	3.3.1.4	3.3.1.4
Perform ACTUATION LOGIC TEST	3.3.1.5	3.3.1.5
Calibrate excore channels to agree with incore	3.3.1.6	3.3.1.6
Perform COT	3.3.1.7	3.3.1.7
Perform COT	3.3.1.8	3.3.1.8
Perform TADOT	3.3.1.9	3.3.1.9
Channel Calibration	3.3.1.10	------------
Channel Calibration (Neutron detectors excluded)	3.3.1.11	3.3.1.10
Channel Calibration (Include resistance temperature detector)	3.3.1.12	-------------
Perform COT	3.3.1.13	3.3.1.13
Perform TADOT	3.3.1.14	3.3.1.11
Perform TADOT (verification of setpoint is not required)	3.3.1.15	-------------
Verify RTS time response	3.3.1.16	-------------
ESFAS Instrumentation	**3.3.2**	**3.3.2**
Channel Check	3.3.2.1	3.3.2.1
Perform Actuation Logic Test	3.3.2.2	3.3.2.7
Perform Actuation Logic Test (continuity may be excluded)	3.3.2.3	-------------
Perform Master Relay Test	3.3.2.4	3.3.2.7
Perform COT	3.3.2.5	3.3.2.2
Perform Slave Relay Test	3.3.2.6	3.3.2.7
Perform TADOT	3.3.2.7	3.3.2.3
Perform TADOT (setpoint verification not required for manual functions)	3.3.2.8	3.3.2.4
Channel Calibration	3.3.2.9	3.3.2.5
ESFAS Time Response	3.3.2.10	3.3.2.7
Verify pressurizer pressure low and steam line pressure low NOT bypassed	-------------	3.3.2.6
PAM Instrumentation	**3.3.3**	**3.3.3**
Channel Check	3.3.3.1	3.3.3.1
Channel Calibration	3.3.3.2	3.3.3.2
Remote Shutdown System	**3.3.4**	-------------
Channel Check	3.3.4.1	-------------
Verify control circuit and transfer switch functional	3.3.4.2	-------------
Channel Calibration	3.3.4.3	-------------
Perform TADOT (reactor trip breakers)	3.3.4.4	-------------
LOP DG Start Instrumentation	**3.3.5**	**3.3.4**
Channel Check	3.3.5.1	-------------
Perform TADOT	3.3.5.2	3.3.4.1
Channel Calibration	3.3.5.3	3.3.4.2
Containment Purge and Exhaust Isolation Instrumentation	**3.3.6**	-------------
Channel Check	3.3.6.1	-------------
Perform Actuation Logic Test	3.3.6.2	-------------

Technical Specification Section Title/Surveillance Description*	TSTF-425	GINNA
Perform Master Relay Test	3.3.6.3	--------------
Perform Actuation Logic Test (applicable to ESFAS instrumentation)	3.3.6.4	--------------
Perform Master Relay Test	3.3.6.5	--------------
Perform COT	3.3.6.6	--------------
Perform Slave Relay Test	3.3.6.7	--------------
Perform TADOT (validation of setpoint not required)	3.3.6.8	--------------
Channel Calibration	3.3.6.9	--------------
Containment Ventilation Isolation Instrumentation	--------------	**3.3.5**
Channel Check	--------------	3.3.5.1
Perform COT	--------------	3.3.5.2
Perform Actuation Logic Test	--------------	3.3.5.3
Channel Calibration	--------------	3.3.5.4
CREFS Actuation Instrumentation	**3.3.7**	--------------
Channel Check	3.3.7.1	--------------
Perform COT	3.3.7.2	--------------
Perform Actuation Logic Test	3.3.7.3	--------------
Perform Master Relay Test	3.3.7.4	--------------
Perform Actuation Logic Test (applicable to ESFAS instrumentation)	3.3.7.5	--------------
Perform Master Relay Test (applicable to ESFAS instrumentation)	3.3.7.6	--------------
Perform Slave Relay Test	3.3.7.7	--------------
Perform TADOT (validation of setpoint not required)	3.3.7.8	--------------
Channel Calibration	3.3.7.9	--------------
CREATS Actuation Instrumentation	--------------	**3.3.6**
Channel Check	--------------	3.3.6.1
Perform COT	--------------	3.3.6.2
Perform TADOT	--------------	3.3.6.3
Channel Calibration	--------------	3.3.6.4
Perform Actuation Logic Test	--------------	3.3.6.5
FBACS Actuation Instrumentation	**3.3.8**	--------------
Channel Check	3.3.8.1	--------------
Perform COT	3.3.8.2	--------------
Perform Actuation Logic Test	3.3.8.3	--------------
Perform TADOT (validation of setpoint not required)	3.3.8.4	--------------
Channel Calibration	3.3.8.5	--------------
BDPS	**3.3.9**	--------------
Channel Check	3.3.9.1	--------------
Perform COT	3.3.9.2	--------------
Channel Calibration	3.3.9.3	--------------
RCS Pressure, Temperature, and Flow DNB Limits	**3.4.1**	**3.4.1**
Verify pressurizer pressure per COLR	3.4.1.1	3.4.1.1
Verify RCS average temperature per COLR	3.4.1.2	3.4.1.2
Verify RCS total flow rate per COLR	3.4.1.3	3.4.1.3
Verify by heat balance that RCS total flow rate per COLR	3.4.1.4	--------------
RCS Minimum Temperature for Criticality	**3.4.2**	**3.4.2**

Technical Specification Section Title/Surveillance Description*	**TSTF-425**	**GINNA**
Verify RCS average temperature in each loop ≥[540]	3.4.2.1	------------
Verify RCS average temperature in each loop ≥[540] Only required if Tave alarm is inoperable and RCS loop Tave <[547]	-------------	3.4.2.2
RCS P/T Limit	**3.4.3**	**3.4.3**
Verify RCS pressure, temperature, heat up and cooldown rates per PTLR	3.4.3.1	3.4.3.1
RCS Loops – MODES 1 and 2	**3.4.4**	-------------
Verify each RCS loop is in operation	3.4.4.1	-------------
RCS Loops – MODES 1 >8.5% RTP	-------------	**3.4.4**
Verify each RCS loop is in operation	-------------	3.4.4.1
RCS Loops – MODES 1 ≤ 8.5% RTP, 2 and 3	-------------	**3.4.5**
Verify required RCS loop is in operation	-------------	3.4.5.1
Verify steam generator secondary side water level	-------------	3.4.5.2
Verify correct breaker alignment and power to required RCP pump	-------------	3.4.5.3
RCS Loops – MODE 3	**3.4.5**	-------------
Verify required RCS loops are in operation	3.4.5.1	-------------
Verify steam generator water level	3.4.5.2	-------------
Verify correct breaker alignment and power to pumps	3.4.5.3	-------------
RCS Loops – MODE 4	**3.4.6**	**3.4.6**
Verify required RCS or RHR loop is in operation	3.4.6.1	3.4.6.1
Verify steam generator water level	3.4.6.2	3.4.6.2
Verify correct breaker alignment and power to pump	3.4.6.3	3.4.6.3
RCS Loops – MODE 5, Loops Filled	**3.4.7**	**3.4.7**
Verify required RHR loop is in operation	3.4.7.1	3.4.7.1
Verify steam generator secondary side water level	3.4.7.2	3.4.7.2
Verify correct breaker alignment and power to required RHR pump	3.4.7.3	3.4.7.3
RCS Loops – MODE 5, Loops Not Filled	**3.4.8**	**3.4.8**
Verify required RHR loop is in operation	3.4.8.1	3.4.8.1
Verify correct breaker alignment and power to RHR pump	3.4.8.2	3.4.8.2
Pressurizer	**3.4.9**	**3.4.9**
Verify pressurizer water level	3.4.9.1	3.4.9.1
Verify capacity of each group of pressurizer heaters	3.4.9.2	3.4.9.2
Verify required pressurizer heaters capable being powered by emergency power	3.4.9.3	---------------
Pressurizer PORVs	**3.4.11**	**3.4.11**
Cycle each block valve	3.4.11.1	3.4.11.1
Cycle each PORV	3.4.11.2	3.4.11.2
Cycle each solenoid air control valve	3.4.11.3	---------------
Verify PORVs and block valves capable being powered by emergency power	3.4.11.4	---------------
LTOP System	**3.4.12**	**3.4.12**
Verify no SI pump capable of injecting	3.4.12.1 3.4.12.2	3.4.12.1
Verify maximum of [one] [HPI] pump capable of injecting	3.4.12.1 3.4.12.2	3.4.12.2
Verify maximum of charging pump capable of injecting	3.4.12.2	3.4.12.2
Verify each accumulator is isolated	3.4.12.3	3.4.12.3

Technical Specification Section Title/Surveillance Description*	**TSTF-425**	**GINNA**
Verify RHR suction valve is open	3.4.12.4	--------------
Verify RCS vent path	3.4.12.5	3.4.12.4
Verify (required) PORV block valve open	3.4.12.6	3.4.12.5
Verify RHR suction valve is locked open and power removed	3.4.12.7	--------------
Perform COT (excluding actuation)	3.4.12.8	3.4.12.6
Verify power removed from each ECCS accumulator MOV	--------------	3.4.12.7
Channel Calibration	3.4.12.9	3.4.12.8
RCS Operational Leakage	**3.4.13**	**3.4.13**
Verify RCS operational leakage within limits	3.4.13.1	3.4.13.1
Verify primary to secondary leakage within limits	3.4.13.2	3.4.13.2
RCS PIV Leakage	**3.4.14**	**3.4.14**
Verify leakage from each RCS PIV	3.4.14.1	3.4.14.1 3.4.14.2
Verify RHR interlock functionality (opening)	3.4.14.2	--------------
Verify RHR interlock functionality (closure)	3.4.14.3	--------------
RCS Leakage Detection Instrumentation	**3.4.15**	**3.4.15**
Channel Check	3.4.15.1	3.4.15.1
Perform COT (atmosphere radioactivity monitor)	3.4.15.2	3.4.15.2
Channel Calibration (containment sump monitor)	3.4.15.3	3.4.15.3
Channel Calibration (containment atmosphere monitor)	3.4.15.4	3.4.15.4
Channel Calibration (containment air cooler flow rate monitor)	3.4.15.5	--------------
RCS Specific Activity	**3.4.16**	**3.4.16**
Verify gross specific activity	3.4.16.1	3.4.16.1
Verify Dose Equivalent I-131	3.4.16.2	3.4.16.2
Determine E	3.4.16.3	3.4.16.3
RCS Loop Isolation Valves	**3.4.17**	--------------
Verify loop isolation valve is open and power removed	3.4.17.1	--------------
RCS Loops – Test Exceptions	**3.4.19**	--------------
Verify thermal power is <P-7	3.4.19.1	--------------
Accumulators	**3.5.1**	**3.5.1**
Verify accumulator isolation valves fully open	3.5.1.1	3.5.1.1
Verify borated water volume	3.5.1.2	3.5.1.2
Verify Nitrogen cover pressure	3.5.1.3	3.5.1.3
Verify boron concentration	3.5.1.4	3.5.1.4
Verify power removed from isolation valve	3.5.1.5	3.5.1.5
ECCS - Operating	**3.5.2**	**3.5.2**
Verify (listed) valves in proper position with power removed	3.5.2.1	3.5.2.1
Verify valves in flow path in the proper position with power removed	3.5.2.2	3.5.2.2
Verify breakers, for each valve listed in 3.5.2.1, in correct position	-------------	3.5.2.3
Verify ECCS piping full of water	3.5.2.3	---------------
Verify ECCS automatic valves actuates to its proper position on an actual or simulated signal	3.5.2.5	3.5.2.5
Verify ECCS pumps starts automatically on an actual or simulated signal	3.5.2.6	3.5.2.6
Verify ECCS throttle valves (listed) in the correct position	3.5.2.7	---------------
Visual inspection of ECCS train	3.5.2.8	3.5.2.7

Technical Specification Section Title/Surveillance Description*	TSTF-425	GINNA
RWST	**3.5.4**	**3.5.4**
Verify RWST borated water temperature	3.5.4.1	---------------
Verify RWST borated water volume	3.5.4.2	3.5.4.1
Verify RWST boron concentration	3.5.4.3	3.5.4.2
Seal Injection Flow	**3.5.5**	-------------
Verify manual seal injection valves adjusted to proper flow	3.5.5.1	-------------
BIT	**3.5.6**	-------------
Verify BIT borated water temperature	3.5.6.1	-------------
Verify BIT borated water volume	3.5.6.2	-------------
Verify BIT boron concentration	3.5.6.3	-------------
Containment Air Locks	**3.6.2**	**3.6.2**
Verify one door can be opened at a time	3.6.2.2	3.6.2.2
Containment Isolation Valves / Boundaries	**3.6.3**	**3.6.3**
Verify purge valve [42 inch] closed	3.6.3.1	-------------
Verify purge valve [8 inch] closed	3.6.3.2	3.6.3.1
Verify manual isolation valves, blind flanges outside containment closed	3.6.3.3	3.6.3.2
Verify isolation times	3.6.3.5	-------------
Cycle each testable check valves through one full cycle	3.6.3.6	-------------
Leakrate purge valves with resilient seals	3.6.3.7	-------------
Verify automatic valves actuate to their correct position on an actual or simulated signal	3.6.3.8	3.6.3.6
Cycle each testable check valves through one full cycle	3.6.3.9	-------------
Verify purge valve is blocked to restrict flow	3.6.3.10	-------------
Containment Pressure	**3.6.4A**	**3.6.4**
Verify containment pressure within limits	3.6.4A.1	3.6.4.1
Containment Temperature	**3.6.5A**	**3.6.5A**
Verify containment average temperature within limits	3.6.5A.1	3.6.5.1
Containment Spray and Cooling Systems / (CS,CRFC and NaOH System)	**3.6.6A**	**3.6.6**
Verify valves in the flow path in the correct position	3.6.6A.1	3.6.6.2
Operate containment cooling system (≥15 minutes)	3.6.6A.2	3.6.6.4
Verify containment cooling train water flow rate(≥700 gpm)	3.6.6A.3	3.6.6.5
Verify containment spray valves actuate to their correct position on an actual or simulated signal	3.6.6A.5	3.6.6.10
Verify containment spray pumps starts on an actual or simulated signal	3.6.6A.6	3.6.6.11
Verify containment cooling train starts on an actual or simulated signal	3.6.6A.7	3.6.6.12
Spray Additive System	**3.6.7**	------------
Verify valves in the flow path in the correct position	3.6.7.1	3.6.6.3
Verify spray tank solution volume	3.6.7.2	3.6.6.7
Verify NaOH concentration	3.6.7.3	3.6.6.8
Verify containment spray additive tank valves actuate to their correct position on an actual or simulated signal	3.6.7.4	3.6.6.13
Verify spray additive flow from each solution flow path	3.6.7.5	3.6.6.14
MSIV / (MSIV and Non-Return Check Valves)	**3.7.2**	**3.7.2**

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Verify MSIV actuate to its isolation position on an actual or simulated signal	3.7.2.2	3.7.2.3
MFIVs and MFRVs, and Associated Bypass Valves	**3.7.3**	**3.7.3**
Verify the isolation time of each MFIV, MFRV and associated bypass valve	3.7.3.2	----------------
Atmosphere Dump (Relief) Valves (ADVs)(ARVs)	**3.7.4**	**3.7.4**
Verify once complete cycle of each ADV(ARV)	3.7.4.1	3.7.4.1
Verify once complete cycle of each ADV (ARV) block valve	3.7.4.2	3.7.4.2
AFW System	**3.7.5**	**3.7.5**
Verify valves in the water and steam flow path in their correct position	3.7.5.1	3.7.5.1
Verify each AFW automatic valve actuates to the isolation position on an actual or simulated signal	3.7.5.3	3.7.5.5
Verify each AFW pump starts automatically on an actual or simulated signal	3.7.5.4	3.7.5.6
Verify each SAFW train can be operated from control room	------------	3.7.5.7
Condensate Storage Tank (CST)	**3.7.6**	**3.7.6**
Verify CST level	3.7.6.1	3.7.6.1
Component Cooling Water System (CCW)	**3.7.7**	**3.7.7**
Verify each CCW valve is in the correct position	3.7.7.1	3.7.7.1
Verify each CCW valve in the flow path actuates to the correct position on an actual or simulated signal	3.7.7.2	-------------
Verify each CCW pump starts automatically on an actual or simulated signal	3.7.7.3	-------------
Service Water System (SWS)	**3.7.8**	**3.7.8**
Verify screen house bay water level and temperatures	-------------	3.7.8.1
Verify each SWS valve is in the correct position	3.7.8.1	3.7.8.2
Verify each SWS valve in the flow path actuates to the correct position on an actual or simulated signal	3.7.8.2	3.7.8.4
Verify each SWS pump starts automatically on an actual or simulated signal	3.7.8.3	3.7.8.5
Verify SW loop header cross-tie valves in correct position	------------	3.7.8.3
Ultimate Heat Sink (UHS)	**3.7.9**	-------------
Verify water level in the UHS	3.7.9.1	-------------
Verify average water temperature in the UHS	3.7.9.2	-------------
Operate each cooling tower fan (≥15 minutes)	3.7.9.3	-------------
Verify cooling fan starts automatically on an actual or simulated signal	3.7.9.4	-------------
Control Room Emergency Filtration System (CREFS)	**3.7.10**	--------------
Operate each CREFS train (≥15 minutes) with the heaters on (≥15 minutes)	3.7.10.1	--------------
Verify each CREF train actuates on an actual or simulated signal	3.7.10.3	--------------
Verify each CREF train maintain a positive pressure	3.7.10.4	--------------
Control Room Emergency Air Temperature Control System (CREATCS)	**3.7.11**	**3.7.9**
Verify the CREATCS removes the assume heat load	3.7.11.1	--------------
Operate CREATS filtration train (≥15 minutes)	--------------	3.7.9.1

Technical Specification Section Title/Surveillance Description*	**TSTF-425**	**GINNA**
Verify each CREATS train actuates on an actual or simulated signal	--------------	3.7.9.3
Auxiliary Building Ventilation System	-------------	**3.7.10**
Verify ABVS in operation	-------------	3.7.10.1
Verify ABVS maintains negative pressure with respect to Aux. Bldg.	-------------	3.7.10.2
ECCS PREACS	**3.7.12**	--------------
Operate each PREAC for ≥10 hours with heaters on or for ≥15 minutes for systems without heaters	3.7.12.1	--------------
Verify each PREAC train actuates on an actual or simulated signal	3.7.12.3	--------------
Verify PREAC can maintained pressure	3.7.12.4	--------------
Verify each ECCS PREAC filter bypass damper closed	3.7.12.5	--------------
FBACS	**3.7.13**	--------------
Operate each FBACS for ≥10 hours with heaters on or for ≥15 minutes for systems without heaters	3.7.13.1	--------------
Verify each FBACS train actuates on an actual or simulated signal	3.7.13.3	--------------
Verify FBACS can maintained pressure	3.7.13.4	--------------
Verify each FBACS filter bypass damper closed	3.7.13.5	--------------
PREACS	**3.7.14**	--------------
Operate each PREAC for ≥10 hours with heaters on or for ≥15 minutes for systems without heaters	3.7.14.1	--------------
Verify each PREAC train actuates on an actual or simulated signal	3.7.14.3	--------------
Verify PREAC can maintained pressure	3.7.14.4	--------------
Verify each ECCS PREAC filter bypass damper closed	3.7.14.5	--------------
Fuel Storage Pool Water Level	**3.7.15**	**3.7.11**
Verify fuel storage pool water level (≥23 feet)	3.7.15.1	3.7.11.1
Spent Fuel Pool Boron Concentration	**3.7.16**	**3.7.12**
Verify spent fuel boron concentration within limits	3.7.16.1	3.7.12.1
Secondary Specific Activity	**3.7.18**	**3.7.14**
Verify specific activity of Dose Equivalent I-131 (≤0.10)	3.7.18.1	3.7.14.1
AC Sources - Operating	**3.8.1**	**3.8.1**
Verify correct breaker alignment	3.8.1.1	3.8.1.1
Verify each diesel starts from standby conditions	3.8.1.2	3.8.1.2
Verify each diesel is synchronized and loaded (for ≥ 60 minutes)	3.8.1.3	3.8.1.3
Verify each day tank contains proper fuel quantity (≥ 220 gallons)	3.8.1.4	3.8.1.4
Check and remove accumulated water	3.8.1.5	--------------
Verify fuel oil transfer operation (from storage tank to day tanks)	3.8.1.6	3.8.1.5
Verify each diesel starts from standby conditions in proper time (≤10 sec)	3.8.1.7	3.8.1.2
Verify transfer of AC power sources (Normal to Alternate)	3.8.1.8	3.8.1.6
Load rejection test (largest post-accident load)	3.8.1.9	3.8.1.7
Verify diesel does not trip and voltage is maintained during and following the load rejection	3.8.1.10	3.8.1.7
Verify diesel performs properly on an actual or simulated loss of offsite power signal	3.8.1.11	3.8.1.9
Verify on an actual or simulated ESF actuation each diesel auto starts from standby conditions	3.8.1.12	3.8.1.9
Verify non critical trips are bypassed	3.8.1.13	3.8.1.8
Verify each diesel operates for greater than 24 hours	3.8.1.14	-------------

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Verify diesel starts and performs properly within 5 minutes of operating for 2 hours a maximum load	3.8.1.15	--------------
Verify diesel synchronizes with offsite power while loaded with emergency loads	3.8.1.16	--------------
Verify an actual or simulated ESF signal overrides a test signal	3.8.1.17	--------------
Verify interval between each sequenced load block	3.8.1.18	--------------
Verify on an actual or simulated loss of offsite power in conjunction with an actual or simulated ESF signal the diesel performs properly	3.8.1.19	--------------
Verify when started simultaneously from standby conditions each diesel performs properly	3.8.1.20	--------------
Diesel Fuel Oil, Lube Oil, and Starting Air	**3.8.3**	**3.8.3**
Verify each fuel oil storage tank volume (gallons)	3.8.3.1	3.8.3.1
Verify lube oil inventory	3.8.3.2	--------------
Verify diesel start receiver pressure	3.8.3.4	--------------
Check and remove accumulated water	3.8.3.5	--------------
DC Sources - Operating	**3.8.4**	**3.8.4**
Verify battery terminal voltage	3.8.4.1	3.8.4.1
Verify battery charger can recharge the battery	3.8.4.2	----------
Verify battery capacity	3.8.4.3	3.8.4.2 3.8.4.3
Battery Parameters	**3.8.6**	**3.8.6**
Verify battery float current	3.8.6.1	---------------
Verify each pilot cell voltage	3.8.6.2	-------------
Verify connected batteries electrolyte level	3.8.6.3	3.8.6.1
Verify average electrolyte temperature (fifth cell of each battery)	-------------	3.8.6.5
Verify specific gravity of pilot cell	-------------	3.8.6.3
Verify specific gravity of each connected cell	--------------	3.8.6.6
Verify each pilot cell temperature	3.8.6.4	3.8.6.4
Verify connected battery cell voltage	3.8.6.5	3.8.6.2
Verify battery capacity	3.8.6.6	3.8.4.3
Inverters – Operating	**3.8.7**	-------------
Verify inverter voltage and alignment to AC buses	3.8.7.1	-------------
AC Instrument Bus Sources – MODES 1, 2, 3 and 4	**-------------**	**3.8.7**
Verify correct static switch alignment	--------------	3.8.7.1
Verify correct Class 1 E CVT alignment	--------------	3.8.7.2
Inverters – Shutdown	**3.8.8**	--------------
Verify inverter voltage and alignment to AC buses	3.8.8.1	--------------
AC Instrument Bus Sources – MODES 5 and 6	**--------------**	**3.8.8**
Verify correct static switch alignment	--------------	3.8.8.1
Verify correct Class 1 E CVT alignmnet	--------------	3.8.8.2
Distribution Systems – Operating	**3.8.9**	**--------------**
Verify correct breaker alignment and voltage of required buses	3.8.9.1	--------------
Distribution Systems – MODES 1, 2, 3 and 4	--------------	**3.8.9**
Verify correct breaker alignment and voltage of required buses	--------------	3.8.9.1
Distribution Systems – Shutdown	**3.8.10**	**--------------**
Verify correct breaker alignment and voltage of required buses	3.8.10.1	--------------
Distribution Systems – MODES 5 and 6	**--------------**	**3.8.10**

Technical Specification Section Title/Surveillance Description*	**TSTF-425**	**GINNA**
Verify correct breaker alignment and voltage of required buses	--------------	3.8.10.1
Boron Concentration	**3.9.1**	**3.9.1**
Verify boron concentration within the COLR	3.9.1.1	3.9.1.1
Unborated Water Source Isolation Valves	**3.9.2**	---------------
Verify each valve that isolates unborated water source is closed	3.9.2.1	--------------
Nuclear Instrumentation	**3.9.3**	**3.9.2**
Channel Check	3.9.3.1	3.9.2.1
Channel Calibration	3.9.3.2	3.9.2.2
Containment Penetrations	**3.9.4**	**3.9.3**
Verify containment penetration in required status	3.9.4.1	3.9.3.1
Verify purge and exhaust valves isolate on an actual or simulated signal	3.9.4.2	3.9.3.2
RHR and Coolant Circulation – High Water Level (>23 Ft)	**3.9.5**	**3.9.4**
Verify one RHR loop in operation circulating reactor coolant	3.9.5.1	3.9.4.1
RHR and Coolant Circulation – Low Water Level (< 23 Ft)	**3.9.6**	**3.9.5**
Verify one RHR loop in operation	3.9.6.1	3.9.5.1
Verify correct breaker alignment and power to operating RHR pump	3.9.6.2	3.9.5.2
Refueling Cavity Water Level	**3.9.7**	**3.9.6**
Verify refueling cavity water level above flange	3.9.7.1	3.9.6.1

* The Technical Specification Section Title/Surveillance Description portion of this attachment is a summary description of the referenced TSTF-425 (NUREG-1431)/Ginna TS Surveillances which is provided for information purposes only and is not intended to be a verbatim description of the TS Surveillances.

ATTACHMENT 6

License Amendment Request

R. E. Ginna Nuclear Power Plant
Docket No. 50-244

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)

<u>Proposed No Significant Hazards Consideration</u>

PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION

Description of Amendment Request: This amendment request involves the adoption of approved changes to the standard technical specifications (STS) for Westinghouse Plants, WOG STS (NUREG-1431), 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 7

License Amendment Request

R. E. Ginna Nuclear Power Plant
Docket No. 50-244

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 accordance with the Surveillance Frequency Control Program

INSERT 2

5.5.17 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 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

INSERT 3

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