Attachment 1 to W3F1-2015-0006

Description and Assessment

(4 Pages Attached)

Attachment 1
Description and Assessment

1.0 DESCRIPTION

The proposed amendment would modify Technical Specifications (TS) by relocating specific surveillance frequencies to a licensee controlled program with the adoption of Technical Specification Task Force (TSTF)-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – Risk-Informed Technical Specification Task Force (RITSTF) Initiative 5." Additionally, the change would add a new program, the Surveillance Frequency Control Program (SFCP), to TS Section 6, Administrative Controls.

The changes are consistent with NRC approved Industry/TSTF Standard Technical Specifications (STS) change TSTF-425, Revision 3, (Rev. 3) (ADAMS Accession No. ML080280275). The Federal Register notice published on July 6, 2009 (74 FR 31996), announced the availability of this TS improvement.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

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

Attachment 2 includes Entergy documentation with regard to PRA technical adequacy consistent with the requirements of Regulatory Guide 1.200, Revision 1 (ADAMS Accession No. ML070240001), 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.

Entergy has concluded that the justifications presented in the TSTF proposal and the Safety Evaluation prepared by the NRC staff are applicable to Waterford 3 and justify this amendment to incorporate the changes to the Waterford 3 TSs.

2.2 Optional Changes and Variations

The proposed amendment is consistent with the STS changes described in TSTF-425, Revision 3, but Entergy proposes variations or deviations from TSTF-425, as identified below and may include differing TS Surveillance numbers:

1. The insert provided in TSTF-425 pertaining to text that describes each frequency relocated to the SFCP has been revised from "The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program" to read "The Surveillance Frequency(ies) is/are based on operating experience, equipment reliability, and plant risk and is/are controlled under the Surveillance Frequency Control Program." This variation is consistent with the inserts proposed by TSTF-425, Rev. 3.

2. The approved programs for Waterford 3 are described in Section 6.0, "Administrative Controls," of the Technical Specifications.

3. Waterford 3 Administrative Controls TS section 6.5.17 references the frequency for TS SR 4.7.6.1.b. This section was revised to reflect that this particular SR is

performed in accordance with the SFCP. This is editorial and reflects what the actual TS SR will state.

4. Attachment 7 provides a cross-reference between the NUREG-1432 surveillances included in TSTF-425 versus the Waterford 3 surveillances included in this amendment. Attachment 7 includes a summary description of the referenced TSTF-425 (NUREG-1432)/ Waterford 3 (WF3) TS surveillances which is provided for information purposes only and is not intended to be a verbatim description of the TS surveillances. This cross-reference highlights the following:
   a. NUREG-1432 surveillances included in TSTF-425 and corresponding Waterford 3 surveillances with plant-specific surveillance numbers,
   b. NUREG-1432 surveillances included in TSTF-425 that are not contained in the Waterford 3 TSs, and
   c. Waterford 3 plant-specific surveillances that are not contained in NUREG-1432 and, therefore, are not included in the TSTF-425 mark-ups.
5. Since the Waterford 3 TSs are custom TSs (CTSs), the applicable surveillance requirements and associated Bases numbers differ from the STSs presented in NUREG-1432 and TSTF-425, but with no impact on the NRC staffs model safety evaluation dated July 6, 2009 (74 FR 31996).
6. For NUREG-1432 surveillances not contained in Waterford 3 TSs, the corresponding mark-ups included in TSTF-425 for these surveillances are not applicable to Waterford 3. This is an administrative deviation from TSTF-425 with no impact on the NRC staffs model safety evaluation dated July 6, 2009 (74 FR 31996).
7. For Waterford 3 plant-specific surveillances not included in the NUREG-1432 markups provided in TSTF-425, Entergy has evaluated these surveillances against TSTF-425, Rev. 3 exclusion criteria. Since these surveillances involve fixed periodic frequencies, relocation of these frequencies is consistent with TSTF-425, Revision 3, and with the NRC's model safety evaluation dated July 6, 2009 (74 FR 31996), including the scope exclusions identified in Section 1.0, "Introduction," of the model safety evaluation. In accordance with TSTF-425, changes to the frequencies for these surveillances would be controlled under the SFCP.

Entergy proposes to relocate all surveillance frequencies except those that reference other approved programs, those that are purely event-driven, those that are event-driven but have a time component for performing the surveillance on a one time basis once the event occurs, or those that are related to specific conditions. Entergy considers the differences listed above to be minor variations or deviations of the type permitted by TSTF-425.

3.0 REGULATORY ANALYSIS

3.1 Applicable Regulatory Requirements/Criteria

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

Entergy has proposed to modify TS by the relocation of specific surveillance frequencies to a licensee controlled document, and controlling changes to surveillance frequencies in accordance with a new program, the SFCP, identified in the administrative controls of TS. The SFCP and TS Section 6.5.18 references NEI 04-10, Revision 1, which provides a risk-informed methodology using plant-specific risk insights and performance data to revise surveillance frequencies within the SFCP. This methodology supports relocating surveillance frequencies from TS to a licensee-controlled document, provided those frequencies are changed in accordance with NEI 04-10, Revision 1.

Entergy's adoption of TSTF-425, Rev. 3, and risk-informed methodology of NEI 04-10, Revision 1, satisfies the key principles of risk-informed decision making applied to changes to TS as delineated in RG 1.177 and RG 1.174, in that:

- The proposed change meets current regulations;
- The proposed change is consistent with defense-in-depth philosophy;
- The proposed change maintains sufficient safety margins;
- Increases in risk resulting from the proposed change are small and consistent with the Commission's Safety Goal Policy Statement; and
- The impact of the proposed change is monitored with performance measurement strategies.

Entergy has concluded that the relationship of the proposed changes to the applicable regulatory requirements presented in the Federal Register notice is applicable to Waterford 3.

3.2 No Significant Hazards Consideration

Entergy has reviewed the proposed no significant hazards consideration determination (NSHC) published in the Federal Register dated July 6, 2009, (74 FR 31996). Entergy has concluded that the proposed NSHC presented in the Federal Register notice is applicable to Waterford 3 and is provided as Attachment 6 to this amendment request, which satisfies the requirements of 10 CFR 50.91(a).

4.0 ENVIRONMENTAL CONSIDERATIONS

Entergy has evaluated the proposed amendment and has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

5.0 REFERENCES

1. TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b," March 18, 2009 (ADAMS Accession Number: ML ML080280275).

2. NRC Notice of Availability of Technical Specification Improvement to Relocate Surveillance Frequencies to Licensee Control - Risk-Informed Technical Specification Task Force (RITSTF) Initiative 5b, Technical Specification Task Force - 425, Revision 3, published on July 6, 2009 (74 FR 31996).

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

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

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

Attachment 2 to W3F1-2015-0006

Documentation of PRA Technical Adequacy

(16 Pages Attached)

Attachment 2
Documentation of PRA Technical Adequacy

TABLE OF CONTENTS

1.0 Purpose

The implementation of the Surveillance Frequency Control Program (also referred to as Technical Specifications Initiative 5b) at the Waterford Steam Electric Station, Unit 3 (Waterford 3) will follow the guidance provided in NEI 04-10, Revision 1 (Reference 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 implemented and documented for future audits by the NRC. If the IDP does not approve the STI revision, the STI value is left unchanged.
- Performance monitoring is conducted as recommended by the IDP. In some cases, no additional monitoring may be necessary beyond that already conducted under the Maintenance Rule. The performance monitoring helps to confirm that no failure mechanisms related to the revised test interval become important enough to alter the information provided for the justification of the interval changes.
- The IDP is responsible for periodic review of performance monitoring results. If it is determined that the time interval between successive performances of a surveillance test is a factor in the unsatisfactory performances of the surveillance, the IDP returns the STI 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 methodology endorses the guidance provided in Regulatory Guide 1.200, Revision 1 (Reference 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 applied to the risk 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 (currently, in RG-1.200 Revision 1 this is just the internal events PRA standard). Provide justification to show that where specific requirements in the standard are not adequately met, it will not unduly impact the results.
   - Identify key assumptions and approximations relevant to the results used in the decision-making process.

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

2.0 Scope

Waterford PRA model Revision 5 is the current and most recent evaluations of the site risk profile for internal event challenges. The Waterford 3 PRA model is a state of the art risk model constructed and documented to meet the ANS/ASME industry standard (ASME/ANS RA-Sa-2009 Reference 4) for technical content. The PRA model quantification process used for the Waterford 3 PRA is based on the linked fault tree (i.e. event tree / fault tree) methodology, which is common in the industry and is a well-known methodology.

Entergy Operations, Inc. employs a detailed, thorough, and comprehensive approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Entergy nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments

and independent peer reviews. The following information describes this approach as it applies to the Entergy/Waterford 3 PRA.

3.0 Analysis

3.1 Waterford 3 PRA Technical Adequacy

The Entergy 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 Entergy procedure EN-DC-151, "PSA Maintenance and Update. Revision 5" (Reference 6). This procedure delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Entergy nuclear generation sites. The Entergy Fleet Nuclear Analysis and PSA program, including EN-DC-151, defines 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:

- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every five years.
- 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.

In addition to these activities, Entergy PSA procedure provides 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 PRA/PSA products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for Entergy 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). This online risk evaluation (EOOS – Equipment Out of Service is the program used at Waterford 3) is used during at power operations and a shutdown version is used to support outage activities.

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 5-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant. Entergy recently completed a model update to the Waterford 3 PRA model in 2013.

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.

3.1.1 Plant Changes Not Yet Incorporated into the PRA Model

A PRA Model Change Request (MCR- Entergy PRA model update tracking database) is created for all issues that are identified that could impact the PRA model. The MCR database includes the identification of those plant changes that could impact the PRA model. As part of the PRA evaluation for each STI change request, a review of open items in the MCR database for Waterford 3 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 nontrivial 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.

3.1.2 Applicability of Peer Review Findings and Observations

As with all U.S commercial nuclear plant PRA models, the primary assessment of model quality and technical adequacy is through the peer review process. The Waterford 3 PRA model has undergone several peer reviews and self-assessments to document the model quality and identify areas for potential improvement. The following past assessments have been documented on Waterford 3 model quality and technical adequacy.

- An independent peer review of the Waterford 3 PRA model was conducted by the Combustion Engineering Owners Group (CEOG) in 2001. This review assessed Revision 2 of the Waterford PRA model. All significant issues identified during this review were addressed in later revisions of the PRA model and model documentation.
- The Waterford 3 PRA underwent a Regulatory Guide 1.200 Revision 1 Peer Review against the ANS/ASME PRA Standard. This peer review was performed using the process defined in Nuclear Energy Institute (NEI) 05-04 (Reference 8). The review was conducted by the Westinghouse Owners Group in August of 2009. The review resulted in 45 Findings (along with 49 Suggestions and 2 Best Practices). The conclusion of the review was that the Waterford 3 PRA model substantially meets the ASME PRA Standard and can be used to support risk-informed applications. The findings and conclusions of this review are contained in LTR-RAM-II-09-039 (Ref 7), "RG 1.200 PRA Peer Review against the ASME PRA Standard Requirements for the Waterford Steam Electric Station, Unit 3 Probabilistic Risk Assessment."
- The Waterford 3 Internal Events PRA model was updated in 2013 and model Revision 5 was completed. The update focused on resolving the Findings from the 2009 peer review and was in large part completed to improve model quality to support the NFPA 805/Fire PRA project. As part of the NFPA 805 License Amendment Request application, a self-assessment was completed to document the status of the 2009 peer review findings. Waterford calculation PRA-W3-05-0051 "Documentation of PRA Model Quality for the Waterford 3 NFPA 805 LAR" (Reference 9) documents this self-assessment.

The current Waterford 3 PRA model is Revision 5 and that is the version assessed in the 2013 self-assessment. PRA-W3-05-0051 concluded that all but 13 of the Findings issued against the PRA model in the 2009 peer review have been explicitly addressed through the updated analysis. Of the 13 open issues, 8 of the unaddressed findings are associated with internal flooding. The remaining five have been judged to have no or minimal impact on risk results.

3.1.3 Consistency with Applicable PRA Standards

As indicated above, the PRA model at Waterford was updated in 2013. Current model documentation is required by Entergy Procedure EN-DC-151 to meet the current industry standard for PRA. The procedure requires all PRA calculations and documents attempt to meet Capability Category II of all Supporting Requirements in ASME/ANS RA-Sa-2009 (Reference 4). Current PRA documents include individual self-assessments documenting how each High Level Requirement (HLR) and Supporting Requirement are met in the PRA product.

The latest full scope peer review was conducted in 2009 and was assessed using Regulatory Guide 1.200 Revision 1 and ANS/ASME RA-Sb-2005, "Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 3). Both of these documents have since been revised and current PRA Standards are judged against Regulatory Guide 1.200 Revision 2 and ASME/ANS RA-Sa-2009. Though the last full scope internal events peer review was against the previous Standard and Regulatory Guide, the current model has been constructed to meet the updated guidance and no significant gaps exist. The Entergy and Waterford PRA staff has significant experience with the current Standard and Regulatory Guide. The self-assessment sections included in recently updated PRA documents use the current Standard. The 2013 self-assessment was a snapshot of the resolutions to the 2009 peer review, but the assessment was made using the updated ANS/ASME Standard and Regulatory Guide 1.200 Revision 2 (Reference 5). The results of that review lead to the supporting requirements (SRs) listed below as not meeting Category II in the PRA model used for this assessment. These SRs are summarized in Table 3-1 along with an assessment of the impact for this application.

The remaining gaps are documented in the MCR database so that they can be tracked and their potential impacts accounted for in applications where appropriate. All current open Findings are judged to have low impact on the PRA model or its ability to support a full range of PRA applications.

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.

TABLE 3-1
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II
OF THE ASME PRA STANDARD

Gap #	Description of Technical Issue	Applicable SR	Current Status/Comment	Importance to Application
Gap #1	Although required by this SR, no evaluation of individual component failure modes, human-induced mechanisms, or other events that could release water into the area were identified. The evaluation assumed that using a guillotine rupture was adequate to not require any specific failures or human-induced mechanisms. This does not meet the intent or specifics of this requirement. Other SRs are also potentially not met when only the use of a guillotine rupture is used. These include: (IF-B3 & IF-D6) Waterford 3 basically characterized all flood sources as catastrophic ruptures but where there are potential spray targets they do evaluate spray impacts.	IF-B2-01	The issues addressed are documented in the MCR database and will be addressed during the next revision to the Waterford Flooding analysis. The current flooding analysis is conservative treating all sources as a guillotine break.	External event (flooding) issue will have little/no impact on STI evaluations.

TABLE 3-1 STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD				
Gap #	**Description of Technical Issue**	**Applicable SR**	**Current Status/Comment**	**Importance to Application**
Gap #2	There are no Engineering calculations available to support some of the statements or assumptions made in the Internal Flooding Analysis. Room dimensions and flood rates are not available to justify depths stated for rooms, some zones credit "air tight" doors as being structurally sound with no justification of door integrity against a static water load. If these calculations exist, they should be either provided in appendices to the report or referenced. On page 89 of the Internal Flooding Report, within the 2nd paragraph, a statement is made that a door is assumed to open out, and that the flood propagation pathway will go through that door. No discussion, calculation or basis, is provided to justify why. On page 215, there is an unsupported assumption that drain failures have a failure probability of 0.1. Need to provide basis for this assumption.	IF-C3c-01	The issues addressed are documented in the MCR database and will be addressed during the next revision to the Waterford Flooding analysis.	External event (flooding) issue will have little/no impact on STI evaluations.

TABLE 3-1
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

Gap #	Description of Technical Issue	Applicable SR	Current Status/Comment	Importance to Application
Gap #3	The Fire Water pump house has been excluded from evaluation on the basis that the failure of the fire pumps will not precipitate a reactor trip. This exclusion needs to be re-visited to determine if an internal flood in the fire water pump house has the potential to initiate a flood/spray event elsewhere in the plant due to spurious fire water valve actuations (e.g. look at potential for spray/submergence on a fire water control panel.), and if this inadvertent actuation could result in the need for a plant shutdown. If this impact has been evaluated, document it.	IF-C7-01	The issues addressed are documented in the MCR database and will be addressed during the next revision to the Waterford Flooding analysis. The issues are essentially a typo level.	External event (flooding) issue will have little/no impact on STI evaluations.
Gap #4	Although Waterford calculates the initiating event frequency for each flood scenario using generic data, and a reduction factor has been inappropriately applied to component rupture failure rates. The analysis states that the generic component failure rates are obtained from EGG-SSRE-9639. However, these failure rates are then reduced by an additional factor to convert them from "spray" failures to "rupture" failures. (The example provided shows a "1/27th" reduction for a 1000 gpm valve failure) The application of the reduction factor is inappropriate. Need to use the "rupture" failure rates without applying the additional reduction factor.	IF-D5a-01	The issues addressed are documented in the MCR database and will be addressed during the next revision to the Waterford Flooding analysis.	External event (flooding) issue will have little/no impact on STI evaluations.

TABLE 3-1
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II
OF THE ASME PRA STANDARD

Gap #	Description of Technical Issue	Applicable SR	Current Status/Comment	Importance to Application
Gap #5	The Internal Flooding report is inconsistent / incorrect in its use of "subsume" versus "screen". For example, in Section 4.2.1.3, the report states that scenarios are "subsumed" but the justification for subsuming the scenarios is based on the justification for "screening" of scenarios (screening is defined in SR IF-D7).	IF-D7-02	The issues addressed are documented in the MCR database and will be addressed during the next revision to the Waterford Flooding analysis. This issue is essentially a documentation issue. The difference between subsume and screen is a technical difference but as applied it is only applied to non-risk significant flooding scenarios.	External event (flooding) issue will have little/no impact on STI evaluations.
Gap #6	The discussion for excluding the condensate polisher building from flooding consideration based on the assumption that the operators would bypass the condensate polisher system in the event of a rupture/leak within the building is inadequate.	IF-D7-01	The issues addressed are documented in the MCR database and will be addressed during the next revision to the Waterford Flooding analysis.	External event (flooding) issue will have little/no impact on STI evaluations.

TABLE 3-1
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

Gap #	Description of Technical Issue	Applicable SR	Current Status/Comment	Importance to Application
Gap #7	For operator actions, only actions outside of the Control Room appear to have been reviewed. Also, no analysis could be found to determine if there were any "unique" operator actions that should be added for internal flooding recoveries, or if the operator actions credited were modified to account for the stress level/timing differences associated with internal flooding scenarios. Of the actions credited in the base PRA model, 4 of the operator actions appear to be removed by a recovery rule file as inaccessible. However, no additional analysis was found to justify why these 4 actions were determined to be inappropriate for internal flooding recovery, or why no other human actions were impacted by the internal flooding scenarios.	IF-E5a-01	The issues addressed are documented in the MCR database and will be addressed during the next revision to the Waterford Flooding analysis. All operator actions outside the control room are reviewed and the operator action is not credited if the flood is on the same elevation as the component being operated locally. Flooding specific operator actions were not specifically identified or addressed	External event (flooding) issue will have little/no impact on STI evaluations.
Gap #8	In general, WSES3 used the standard quantification processes from section 4.5.8 of the standard. However, WSES3 did not propagate the numerical uncertainties as part of the quantification. WSES3 needs to redo the Internal Flooding Quantification and include the propagation of the numerical uncertainties and provide the mean and ERF factors for the resultant CDFs.	IF-E6-01	The issues addressed are documented in the MCR database and will be addressed during the next revision to the Waterford Flooding analysis.	External event (flooding) issue will have little/no impact on STI evaluations.

TABLE 3-1
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II
OF THE ASME PRA STANDARD

Gap #	Description of Technical Issue	Applicable SR	Current Status/Comment	Importance to Application
Gap #9	In the Accident Sequence Notebook, assumption 2.20 reads: ...for SGTRs, failure of ADV to close after opening is not included due to block valves upstream of the ADV that could be closed by the operator. It is not clear if after not modeling the failure to ADV to close, if the closure of the block valves by the operator has been modeled. If it is not modeled, the review team believes that it should be, so as to not lose the dependency that this operator action might have on other operator actions.	AS-A7-02	The Accident Sequence Notebook was updated but the requested action to close the ADVs was not added to the model. The change/update would likely reduce CDF and LERF, but SGTR contributions are low (less than 2%) and the model is conservative as-is. Based on the current treatment a finding would not likely be issued for this same item because the treatment is more clearly documented It is considered still open only because the re requested method of closure was no adopted.	Impact on quantified results is both minor and conservative. The Waterford 3 PRA staff intentionally chose not to address as the action requested would have little impact on risk results. This issue will not impact STI evaluations.
Gap #10	ISLOCA – low pressure LPSI and HPSI line contain two check valves in series. The failure rate of the check valves need to be treated as conditional, rather than independent. Additionally need to address small ruptures in the LPSI MOVs. At present only large leakage is considered.	IE-C12-01	This finding is associated with the inclusion of State of Knowledge Correlation (SOKC). The increase in probabilities due to SOKC would be minor per industry guidance. This is insignificant in the overall PRA because the conditional failure rate of both check valves would be about 2E-7. Additionally, a review of NUREG/CR-6928 shows that small ruptures are defined as 1 to 50 gpm. Leaks of this size are not considered sufficient to meet the classification for ISLOCA.	This impact of this finding is only associated with the uncertainty evaluation of ISLOCA and it is such a small contributor that the overall impact is negligible (and it does not impact STI evaluations).

TABLE 3-1 STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD				
Gap #	**Description of Technical Issue**	**Applicable SR**	**Current Status/Comment**	**Importance to Application**
Gap #11	Tables 4.5.8-2 d and e of the ASME Standard include requirements such as documenting a review of a sample of the significant accident sequences/cutsets, comparing the overall LERF and contributors to similar plants, reviewing a sample of non-significant cutsets, identifying significant contributors (such as initiating events, equipment failures, CCFs, and HFEs), review of component importance measures, and evaluating the overall LERF uncertainty intervals. The significant LERF contributors are presented in Section 4.3 of the LERF Report, a comparison to a similar plant is presented in Section 4.5, and parametric uncertainty was performed in Appendix E, but the other requirements have not been documented.	LE-F3-01	The finding has been partially addressed. Every element listed in the finding has not been completed and documented. No review of importance measures is documented. Besides the review of importance measures, all listed requirements are included in the current model documentation.	The finding is not fully addressed but the noted deficiency is minor and does not impact the use of the LERF model.
Gap #12	Need to add a discussion of what the criteria for CCF considerations are (which types of components were looked at, were inter- and intra- system CCFs considered, etc. If the component types were determined based off of a list from a Reference, provide this information and a pointer to the reference document/methodology.	SY-C2-01	This is a documentation issue. The CCF (grouping, inter and intersystem considerations) is adequately evaluated. There is just a gap between the current documentation and the requirements in the Standard. The issue is in the MCR database and will be addressed during the next revision.	No impact on quantified results (a documentation limitation only). Will not impact STI evaluations.

TABLE 3-1
STATUS OF IDENTIFIED GAPS TO CAPABILITY CATEGORY II OF THE ASME PRA STANDARD

Gap #	Description of Technical Issue	Applicable SR	Current Status/Comment	Importance to Application
Gap #13	HPSI system has an installed spare that can be aligned to either system. Coincident unavailability due to maintenance for redundant equipment is possible (spare pump OOS for extended periods and could be OOS with another pump). Need to specifically address this possibility. This may also be true for charging pumps	SY-A18a-01	Coincident maintenance is associated with repetitive, planned maintenance. Therefore, the types of activities mentioned in the finding are not consistent with the definition of concurrent maintenance in the standard. The Plant Specific Failure Data Development analysis (PSA-WF3-01-DA-01) Appendix 10 of the WF3 Data Analysis (PRA-W3-01-001S05, Rev. 1) documents the inclusion of all planned concurrent maintenance (including installed spares). This remains 'not addressed' due to documentation. The coincident unavailability is included in the model, however the documentation does not fully explain the process used to consider/model events.	This is a documentation issue and has no impact on results and will not impact STI evaluations.

3.2 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, 2.2.2 and 2.2.3 above will be documented and included in the results of the risk analysis that goes to the IDP. This will include, for each STI change assessment, a review of identified sources of uncertainty that were developed for Waterford 3 based on the guidance in EPRI TR-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," (Reference 10).

3.3 External Events Considerations

The NEI 04-10 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 Waterford 3 Individual Plant Examination of External Events (IPEEE) submittal in response to the NRC IPEEE Program (Generic Letter 88-20, Supplement 4) (Reference 11). 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. No Waterford 3 PRA model or applications associated with external hazards seismic, high wind, external flooding, exists that could provide quantitative insights to support this STI effort.

Waterford 3 does have a state of the art Fire PRA model (peer reviewed against the current revision of the PRA Standard and Revision 2 of Regulatory Guide 1.200). This Fire PRA model is based on the internal events model and is also of sufficient quality to support PRA applications. Any STI related parameter changes evaluated by the internal events model can also be evaluated using the Fire model.

As stated earlier, the NEI 04-10 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. The Fire PRA model will be exercised to obtain quantitative fire risk insights when a qualitative or a bounding analysis is not deemed sufficient, but refinements may need to be made on a case-by-case basis. This approach is consistent with the accepted NEI 04-10 methodology.

4.0 Conclusions

The Waterford 3 PRA technical adequacy and capability evaluations and the maintenance and update processes described above provide a robust basis for concluding that the Waterford 3 PRA is suitable for use in risk-informed applications. This includes the proposed implementation of a Surveillance Frequency Control Program. Also, in addition to the standard set of sensitivity studies required per the

NEI 04-10 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.

5.0 References

1. NEI 04-10, Revision 1. Risk-Informed Technical Specifications Initiative 5b Risk-Informed Method for Control of Surveillance Frequencies
2. Regulatory Guide 1.200, Revision 1. An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."
3. ANS/ASME RA-Sb-2005 - American Society of Mechanical Engineers, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, (ASME RA-S-2002), Addenda RA-Sb-2005, December 2005.
4. ASME/ANS RA-Sa-2009 - American Society of Mechanical Engineers, Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, February 2009
5. Regulatory Guide 1.200, Revision 2. An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."
6. EN-DC-151 - PSA Maintenance and Update. Revision 5 - Entergy Procedure
7. LTR-RAM-II-09-039, RG 1.200 PRA Peer Review against the ASME PRA Standard Requirements for the Waterford Steam Electric Station, Unit 3 Probabilistic Risk Assessment.
8. (NEI) 05-04 - : Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard, November 2008
9. PRA-W3-05-0051, Revision 0 (documented in Waterford 3 EC 50448). Documentation of PRA Model Quality for the Waterford 3 NFPA 805 LAR, December 2013.
10. EPRI TR-1016737, Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty, December
11. 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

Attachment 3 to W3F1-2015-0006

Proposed Technical Specification Changes

(115 Pages Attached)

INSERT 1a

in accordance with the Surveillance Frequency Control Program

or

INSERT 1b

In accordance with the Surveillance Frequency Control Program

INSERT 2a

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

or

INSERT 2b

The Surveillance Frequencies are based on operating experience, equipment reliability, and plant risk and are controlled under the Surveillance Frequency Control Program.

INSERT 3

6.5.18 <u>Surveillance Frequency Control Program</u>

This program provides controls for 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 those Surveillance Requirements for which the Frequency is controlled by the program.

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

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

DEFINITIONS

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

SOFTWARE

1.30 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

1.31 Definition 1.31 has been deleted.

SOURCE CHECK

1.32 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

~~STAGGERED TEST BASIS~~

1.33 ~~A STAGGERED TEST BASIS shall consist of:~~ Definition 1.33 has been deleted.

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

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

THERMAL POWER

1.34 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.35 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

TABLE 1.1

FREQUENCY NOTATION

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - ANY CEA WITHDRAWN

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the COLR.

APPLICABILITY: MODES 1, 2*, 3, 4, and 5 with any CEA fully or partially withdrawn.

ACTION:

With the SHUTDOWN MARGIN less than that specified in the COLR, immediately initiate boration to restore SHUTDOWN MARGIN to within limit.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 With any CEA fully or partially withdrawn, the SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the COLR:

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

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, ~~at least once per 12 hours~~ by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

Add INSERT 1a

---

* See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the ~~CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.~~

e. When in MODE 3, 4, or 5, ~~at least once per 24 hours~~ by consideration of at least the following factors:

Add INSERT 1a

1. Reactor Coolant System boron concentration,

2. CEA position,

3. Reactor Coolant System average temperature,

4. Fuel burnup based on gross thermal energy generation,

5. Xenon concentration, and

6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1.0% delta k/k ~~at least once per 31 Effective Full Power Days (EFPD)~~. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPDs after each fuel loading.

Add INSERT 1a

SHUTDOWN MARGIN - ALL CEAS FULLY INSERTED

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the COLR.

APPLICABILITY: MODES 3, 4 and 5 with all CEAs fully inserted.

ACTION:

With the SHUTDOWN MARGIN less than that specified in the COLR, immediately initiate boration to restore SHUTDOWN MARGIN to within limit.

SURVEILLANCE REQUIREMENTS

4.1.1.2 With all CEAs fully inserted, the SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the COLR, ~~at least once per 24 hours~~ by consideration of the following factors:

Add INSERT 1a

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

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{cold}) shall be greater than or equal to 533°F.

APPLICABILITY: MODES 1 and 2#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{cold}) less than 533°F, restore T_{cold} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{cold}) shall be determined to be greater than or equal to 533°F ~~at least once per 12 hours~~.

Add INSERT 1a

#With K_{eff} greater than or equal to 1.0.

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid makeup tank via either a boric acid makeup pump or a gravity feed connection and any charging pump to the Reactor Coolant System if the boric acid makeup tank in Specification 3.1.2.7a. is OPERABLE, or

b. The flow path from the refueling water storage pool via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System if the refueling water storage pool in Specification 3.1.2.7b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.*

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE ~~at least once per 31 days~~ by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

Add INSERT 1a

* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two boron injection flow paths to the RCS via the charging pumps shall be OPERABLE. The following flow paths may be used:

a. With the contents of either boric acid makeup tank in accordance with Figure 3.1-1, the following flow paths shall be OPERABLE:

   1. One flow path from an acceptable boric acid makeup tank via its boric acid makeup pump; and

   2. One flow path from an acceptable boric acid makeup tank via its gravity feed valve; or

b. With the combined contents of both boric acid makeup tanks in accordance with Figure 3.1-2, both of the following flow paths shall be OPERABLE:

   1. One flow path consisting of both boric acid makeup pumps, and

   2. One flow path consisting of both gravity feed valves.

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.1 or 3.1.1.2, whichever is applicable, within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

Add INSERT 1a

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

b. ~~At least once per 18 months during shutdown~~ by verifying that each automatic valve in the flow path actuates to its correct position on an SIAS test signal.

c. ~~At least once per 18 months~~ by verifying that the flow path required by Specification 3.1.2.2a.1 and 3.1.2.2a.2 delivers at least 40 gpm to the Reactor Coolant System.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two independent charging pumps shall be OPERABLE.

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

ACTION:

~~With only one charging pump OPERABLE, restore at least two charging pumps to~~ OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.1 or 3.1.1.2, whichever is applicable, within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 Each required charging pump shall be demonstrated OPERABLE ~~at least once every 18 months~~ by verifying that each charging pump starts in response to an SIAS test signal.

Add INSERT 1a

BORIC ACID MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a. shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2a. is OPERABLE.

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

ACTION:

With one boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a. inoperable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.1 or 3.1.1.2, whichever is applicable, restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each required boric acid makeup pump shall be demonstrated OPERABLE ~~at least once every 18 months~~ by verifying that each boric acid makeup pump starts in response to an SIAS test signal.

Add INSERT 1a

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid makeup tank with a boron concentration between 4900 ppm and 6125 ppm and a minimum borated water volume of 36% indicated level.

b. The refueling water storage pool (RWSP) with:

1. A minimum contained borated water volume of 12% indicated level, and

2. A minimum boron concentration of 2050 ppm.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes. *

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

Add INSERT 1b

a. ~~At least once per 24 hours~~ when the Reactor Auxiliary Building air temperature is less than 55°F by verifying the boric acid makeup tank solution is greater than or equal to 60°F (when it is the source of borated water).

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

1. Verifying the boron concentration of the water, and

2. Verifying the contained borated water volume of the tank.

* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. At least one of the following sources:

1) One boric acid makeup tank, with the tank contents in accordance with Figure 3.1-1, or

2) Two boric acid makeup tanks, with the combined contents of the tanks in accordance with Figure 3.1-2, and

b. The refueling water storage pool in accordance with Specification 3.5.4.

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

ACTION:

a. With the above required boric acid makeup tank(s) inoperable, restore the tank(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.1 or 3.1.1.2, whichever is applicable; restore the above required boric acid makeup tank(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage pool inoperable, restore the pool to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

a. ~~At least once per 24 hours by~~ verifying the boric acid makeup tank solution temperature is greater than or equal to 60°F when the Reactor Auxiliary Building air temperature is less than 55°F.

Add INSERT 1b

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

1. Verifying the boron concentration in the water, and

2. Verifying the contained borated water volume of the water source.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.9.1 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 from MODE 2.

Add INSERT 1a

4.1.2.9.2 Each required boron dilution alarm shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK ~~at least once per 12 hours~~, a CHANNEL FUNCTIONAL TEST ~~at least once per 31 days~~, and a CHANNEL CALIBRATION ~~at least once per 18 months~~.

4.1.2.9.3 If the primary makeup water flow path to the Reactor Coolant System is isolated to fulfill 3.1.2.9.b, the required primary makeup water flow path to the Reactor Coolant System shall be verified to be isolated by either locked closed manual valves, deactivated automatic valves secured in the isolation position, or by power being removed from all charging pumps, at least once per 24 hours.

4.1.2.9.4 The requirements of Specification 3.1.2.9.a.2 or 3.1.2.9.b.2 shall be verified ~~at least once per 24 hours~~.

4.1.2.9.5 Each required boron dilution alarm setpoint shall be adjusted to less than or equal to the existing neutron flux (cps) multiplied by the value specified in the COLR, at the frequencies specified in the COLR.

Add INSERT 1a

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group ~~at least once per 12 hours~~ except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction ~~at least once per 92 days~~.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5 inches, and

c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

a. Restore the inoperable position indicator channel to OPERABLE status, or

b. Be in at least HOT STANDBY, or

c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specifications 3.1.3.1 and 3.1.3.6. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out" limit.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5 inches of each other ~~at least once per 12 hours~~.

Add INSERT 1a

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each CEA not fully inserted.

APPLICABILITY: MODES 3*, 4*, and 5*.

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST ~~at least once per 18 months~~. The provisions of Specification 4.0.4 are not applicable for performance of this surveillance testing.

Add INSERT 1a

*With the reactor trip breakers in the closed position.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to greater than or equal to 145 inches.

APPLICABILITY: MODES 1 and 2*#**.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 145 inches withdrawn, within 1 hour either:

a. Withdraw the CEA to greater than or equal to 145 inches, or

b. Declare the CEA inoperable and determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to greater than or equal to 145 inches withdrawn:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups or group P during an approach to reactor criticality, and

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

Add INSERT 1b

*See Special Test Exception 3.10.2.

#With Keff greater than or equal to 1.0.

**Except for surveillance testing pursuant to Specification 4.1.3.1.2.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With the regulating CEA groups or group P CEAs inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per calendar year, either:

1. Restore the regulating CEA groups or group P CEAs to within the Long Term Steady State Insertion Limits within two hours, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group and CEA group P shall be determined to be within the Transient Insertion Limits ~~at least once per 12 hours~~ except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups or CEA group P are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined ~~at least once per 24 hours~~.

Add INSERT 1a

Add INSERT 1a

LIMITING CONDITION FOR OPERATION

(COLSS) or, with the COLSS out of service, by verifying ~~at least once per 2 hours~~ that the linear heat rate, as indicated on any OPERABLE Local Power Density channel, is within the limits specified in the COLR.

4.2.1.3 ~~At least once per 31 days~~, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on kW/ft.

Add INSERT 1b

POWER DISTRIBUTION LIMITS

3/4.2.2 PLANAR RADIAL PEAKING FACTORS - F_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With a F^m_{xy} exceeding a corresponding F^c_{xy}, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to planar radial peaking by a factor equivalent to greater than or equal to F^m_{xy}/F^c_{xy} and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[F^m_{xy}/F^c_{xy}) - 1.0]$ x 100% is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) or

c. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}), used in the COLSS and CPC at the following intervals:

a. ~~After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and~~

b. ~~At least once per 31 effective full power days (EFPD).~~

Add INSERT 1b

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. ~~Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.~~

b. Calculating the tilt ~~at least once per 12 hours~~ when the COLSS is inoperable.

Add INSERT 1a

c. Verifying ~~at least once per 31 days,~~ that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors ~~at least once per 31 days~~ to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

INSERT 1a

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying ~~at least once per 2 hours~~ that the DNBR, as indicated on any OPERABLE DNBR channel, is within the limit specified in the COLR.

4.2.4.3 ~~At least once per 31 days~~, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

Add INSERT 1b

POWER DISTRIBUTION LIMITS

3/4.2.5 RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 148.0 x 10^6 lbm/h.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be greater than or equal to the above limit ~~at least once per 12 hours~~.

Add INSERT 1a

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The reactor coolant cold leg temperature (T_c) shall be maintained between 536°F and 549°F.*

APPLICABILITY: MODE 1 above 30% of RATED THERMAL POWER.

ACTION:

With the reactor coolant cold leg temperature exceeding its limit, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The reactor coolant cold leg temperature ~~shall be determined~~ to be within its limit ~~at least once per 12 hours~~.

Add INSERT 1a

*Following a reactor power cutback in which (1) Regulating Groups 5 and/or 6 are dropped or (2) Regulating Groups 5 and/or 6 are dropped and the remaining Regulating Groups (Groups 1, 2, 3, and 4) are sequentially inserted, the upper limit on T_c may increase to 559°F for up to 30 minutes.

3/4.2.7 AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The AXIAL SHAPE INDEX (ASI) shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With the AXIAL SHAPE INDEX outside the limits specified in the COLR, restore the AXIAL SHAPE INDEX to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The AXIAL SHAPE INDEX shall be determined to be within its limit ~~at least once per 12 hours~~ using the COLSS or any OPERABLE Core Protection Calculator channel.

Add INSERT 1a

*See Special Test Exception 3.10.2.

3/4.2.8 PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The steady-state pressurizer pressure shall be maintained between 2125 psia and 2275 psia.

APPLICABILITY: MODE 1

ACTION:

With the steady-state pressurizer pressure outside its above limits, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The steady-state pressurizer pressure shall be determined to be within its limit ~~at least once per 12 hours.~~

Add INSERT 1a

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE ~~at least once per 18 months~~ during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

Add INSERT 1a

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

4.3.1.4 The isolation characteristics of each CEA isolation amplifier and each optical isolator for CEA Calculator to Core Protection Calculator data transfer shall be verified ~~at least once per 18 months~~ during the shutdown per the following tests:

a. For the CEA position isolation amplifiers:

1. With 120 volts AC (60 Hz) applied for at least 30 seconds across the output, the reading on the input does not exceed 0.015 volts DC.

2. With 120 volts AC (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 15.0 volts DC.

b. For the optical isolators: Verify that the input to output insulation resistance is greater than 10 megohms when tested using a megohmmeter on the 500 volt DC range.

4.3.1.5 The Core Protection Calculator System and the Control Element Assembly Calculator System shall be determined OPERABLE ~~at least once per 12 hours~~ by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

Add INSERT 1a

4.3.1.6 The Core Protection Calculator System shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a High CPC Cabinet Temperature alarm.

AMENDMENT NO.

Replace each marked through surveillance frequency with 'SFCP'

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	~~R~~ and S/U(1)	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	~~S~~	~~D~~(2,4),~~M~~(3,4), ~~Q~~(4)	~~Q~~	1, 2
3. Logarithmic Power Level - High	~~S~~	~~R~~(4)	~~Q~~ and S/U(1)	2#, 3, 4, 5
4. Pressurizer Pressure - High	~~S~~	~~R~~	~~Q~~	1, 2
5. Pressurizer Pressure - Low	~~S~~	~~R~~	~~Q~~	1, 2
6. Containment Pressure - High	~~S~~	~~R~~	~~Q~~	1, 2
7. Steam Generator Pressure - Low	~~S~~	~~R~~	~~Q~~	1, 2
8. Steam Generator Level - Low	~~S~~	~~R~~	~~Q~~	1, 2
9. Local Power Density - High	~~S~~	~~D~~(2,4), ~~R~~(4,5)	~~Q~~, ~~R~~(6)	1, 2
10. DNBR - Low	~~S~~	~~S~~(7), ~~D~~(2,4), ~~M~~(8), ~~R~~(4,5)	~~Q~~, ~~R~~(6)	1, 2
11. DELETED				
12. Reactor Protection System Logic	N.A.	N.A.	~~Q~~(11) and S/U(1)	1, 2, 3*, 4*, 5*

TABLE (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	~~Q~~ (10,11), S/U(1)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	~~S~~	~~D~~(2,4),~~R~~(4,5)	~~Q~~(9),~~R~~(6)	1, 2
15. CEA Calculators	~~S~~	~~R~~	~~Q~~,~~R~~(6)	1, 2
16. Reactor Coolant Flow - Low	~~S~~	~~R~~	~~Q~~	1, 2

Replace each marked through surveillance frequency with 'SFCP'.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(3) Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine or verify acceptable values for the shape annealing matrix elements used in the Core Protection Calculators.

(6) This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow co-efficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty is included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculatio[ns.]

Add INSERT 1b

(9) The ~~quarterly~~ CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC.

(10) ~~At least once per 18 months~~ and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage trip function and the shunt trip function.

(11) The ~~quarterly~~ CHANNEL FUNCTIONAL TEST shall be scheduled and performed such that the Reactor Trip Breakers (RTBs) are tested at least every 6 weeks to accommodate the appropriate vendor recommended interval for cycling of each RTB.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

1

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE ~~at least once per 18 months~~ during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

Add INSERT 1a

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

Replace each marked through surveillance frequency with 'SFCP'.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTAION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	~~R~~	1, 2, 3, 4
b. Containment Pressure - High	~~S~~	~~R~~	~~Q~~	1, 2, 3
c. Pressurizer Pressure - Low	~~S~~	~~R~~	~~Q~~	1, 2, 3
d. Automatic Actuation Logic (except subgroup relays)	N.A.	N.A.	~~Q~~(2)	1, 2, 3
Actuation Subgroup Relays	N.A.	N.A.	~~M~~(3) (6)	1, 2, 3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	~~R~~	1, 2, 3, 4
b. Containment Pressure -- High - High	~~S~~	~~R~~	~~Q~~	1, 2, 3
c. Automatic Actuation Logic (except subgroup relays)	N.A.	N.A.	~~Q~~(2)	1, 2, 3
Actuation Subgroup Relays	N.A.	N.A.	~~M~~(1) (3)	1, 2, 3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	~~R~~	1, 2, 3, 4
b. Containment Pressure - High	~~S~~	~~R~~	~~Q~~	1, 2, 3
c. Pressurizer Pressure - Low	~~S~~	~~R~~	~~Q~~	1, 2, 3
d. Automatic Actuation Logic (except subgroup relays)	N.A.	N.A.	~~Q~~(2)	1, 2, 3
Actuation Subgroup Relays	N.A.	N.A.	~~M~~(1) (3)	1, 2, 3
4. MAIN STEAM LINE ISOLATION				
a. Manual (Trip Buttons)	N.A.	N.A.	~~R~~	1, 2, 3
b. Steam Generator Pressure - Low	~~S~~	~~R~~	~~Q~~	1, 2, 3
c. Containment Pressure - High	~~S~~	~~R~~	~~Q~~	1, 2, 3
d. Automatic Actuation Logic (except subgroup relays)	N.A.	N.A.	~~Q~~(2)	1, 2, 3
Actuation Subgroup Relays	N.A.	N.A.	~~M~~(1) (3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. SAFETY INJECTION SYSTEM RECIRCULATION (RAS)				
a. Manual RAS (Trip Buttons)	N.A.	N.A.	~~R~~	1, 2, 3, 4
b. Refueling Water Storage Pool - Low	~~S~~	~~R~~	~~Q~~	1, 2, 3, 4
c. Automatic Actuation Logic (except subgroup relays)	N.A.	N.A.	~~Q~~(2)	1, 2, 3, 4
Actuation Subgroup Relays	N.A.	N.A.	~~M~~(1) (3)	1, 2, 3, 4
6. LOSS OF POWER (LOV)				
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	~~R~~	~~D~~(4)	1, 2, 3
b. 480 V Emergency Bus Undervoltage (Loss of Voltage)	N.A.	~~R~~	~~D~~(4)	1, 2, 3
c. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	~~R~~	~~D~~(4)	1, 2, 3

Replace each marked through surveillance frequency with 'SFCP'.

TABLE 4.3.-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	~~R~~	1, 2, 3
b. SG Level (1/2)-Low and ΔP (1/2) - High	~~S~~	~~R~~	~~Q~~	1, 2, 3
c. SG Level (1/2) - Low and No Pressure - Low Trip (1/2)	~~S~~	~~R~~	~~Q~~	1, 2, 3
d. Automatic Actuation Logic (except subgroup relays)	N.A.	N.A.	~~Q~~(2)	1, 2, 3
Actuation Subgroup Relays	N.A.	N.A.	~~M~~(1) (3)	1, 2, 3
e. Control Valve Logic (Wide Range SG Level - Low)	~~S~~	~~R~~	~~SA~~(5)	1, 2, 3

Replace each marked through surveillance frequency with 'SFCP'.

TABLE NOTATION

(1) Each train or logic channel shall be tested ~~at least every 62 days on a STAGGERED TEST BASIS~~.

(2) Testing of Automatic Actuation Logic shall include energization/deenergization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays K109, K114, K202, K301, K305, K308 and K313 are exempt from testing during power operation but shall be tested ~~at least once per 18 months~~ and during each COLD SHUTDOWN condition unless tested ~~within the previous 62 days~~.

Add INSERT 1a

(4) Using installed test switches.

(5) To be performed during each COLD SHUTDOWN if not performed in the previous 6 months.

(6) Each train shall be tested, with the exemption of relays, K110, K410 and K412, ~~at least every 62 days on a STAGGERED TEST BASIS.~~ Relays K110, K410 and K412 shall be tested ~~at least every 62 days but will be exempt from the STAGGERED TEST BASIS~~.

Replace each marked through surveillance frequency with 'SFCP'.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Deleted				
b. Containment - Purge & Exhaust Isolation	~~S~~	~~R~~	~~Q~~	1, 2, 3, 4 & **
2. PROCESS MONITORS				
a. DELETED				
b. Control Room Intake Monitors	~~S~~	~~R~~	~~Q~~	ALL MODES & ***
c. Steam Generator Blowdown	~~S~~	~~R~~	~~Q~~	1, 2, 3, & 4
d. Component Cooling Water Monitors A&B	~~S~~	~~R~~	~~Q~~	ALL MODES
e. Component Cooling Water Monitor A/B	~~S~~	~~R~~	~~Q~~	1, 2, 3, & 4

*Deleted
**During CORE ALTERATIONS or load movements with or over irradiated fuel within the containment.
***During load movements with or over irradiated fuel.

Replace each marked through surveillance frequency with 'SFCP'.

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. EFFLUENT ACCIDENT MONITORS				
a. Containment High Range	~~S~~	~~R~~	~~Q~~	1, 2, 3, & 4
b. Plant Stack High Range	~~S~~	~~R~~	~~Q~~	1, 2, 3, & 4
c. Condenser Vacuum Pump High Range	~~S~~	~~R~~	~~Q~~	1, 2, 3, & 4
d. Fuel Handling Building Exhaust High Range	~~S~~	~~R~~	~~Q~~	1*, 2*, 3*, & 4*
e. Main Steam Line High Range	~~S~~	~~R~~	~~Q~~	1, 2, 3, & 4

*With irradiated fuel in the storage pool.

TABLE 4.3-6

REMOTE SHUTDOWN INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Neutron Flux	~~M~~	~~R~~*
2. Reactor Trip Breaker Indication	~~M~~	N.A.
3. Reactor Coolant Temperature-Cold Leg (T_{Cold})	~~M~~	~~R~~
4. Reactor Coolant Temperature - Hot Leg (T_{Hot})	~~M~~	~~R~~
5. Pressurizer Pressure	~~M~~	~~R~~
6. Pressurizer Level	~~M~~	~~R~~
7. Steam Generator Level	~~M~~	~~R~~
8. Steam Generator Pressure	~~M~~	~~R~~
9. Shutdown Cooling Flow Rate	~~M~~	~~R~~
10. Emergency Feedwater Flow Rate	~~M~~	~~R~~
11. Condensate Storage Pool Level	~~M~~	~~R~~

Replace each marked through surveillance frequency with 'SFCP'.

*Neutron detector may be excluded from CHANNEL CALIBRATION.

AMENDMENT NO.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure (Wide Range)	~~M~~	~~R~~
2. Containment Pressure (Wide Wide Range)	~~M~~	~~R~~
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range	~~M~~	~~R~~
4. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	~~M~~	~~R~~
5. Reactor Coolant Pressure - Wide Range	~~M~~	~~R~~
6. Pressurizer Water Level	~~M~~	~~R~~
7. Steam Generator Water Level - Narrow Range	~~M~~	~~R~~
8. Steam Generator Water Level - Wide Range	~~M~~	~~R~~
9. Containment Water Level (Wide Range)	~~M~~	~~R~~
10. Core Exit Thermocouples	~~M~~	~~R~~
11. Containment Isolation Valve Position	~~M~~	~~R~~
12. Condensate Storage Pool Level	~~M~~	~~R~~
13. Reactor Vessel Level Monitoring System	~~M~~	~~R~~
14. Log Power Indication (Neutron Flux)	~~M~~	~~R~~

Replace each marked through surveillance frequency with 'SFCP'.

INSTRUMENTATION

CHEMICAL DETECTION SYSTEMS

CHLORINE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.7.1 Two independent chlorine detection systems, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 2 ppm, shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

a. With one chlorine detection system inoperable, restore the inoperable detection system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room ventilation system in the isolate mode of operation.

b. With no chlorine detection system OPERABLE, within 1 hour initiate and maintain operation of the control room ventilation system in the isolate mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each chlorine detection system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK ~~at least once per 12 hours~~ and a CHANNEL CALIBRATION ~~at least once per 31 days~~.

Add INSERT 1a

Correction letter of 3-17-2000

INSTRUMENTATION

CHEMICAL DETECTION SYSTEMS

BROAD RANGE GAS DETECTION

LIMITING CONDITION FOR OPERATION

3.3.3.7.3 Two independent broad range gas detection systems shall be OPERABLE ** with their alarm/trip setpoints adjusted to actuate at the lowest achievable Immediately Dangerous to Life or Health gas concentration level of detectable toxic gases* providing reliable operation.

APPLICABILITY: All MODES.

ACTION :

a. With one broad range gas detection system inoperable, restore the inoperable detection system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room ventilation system in the isolate mode of operation.

b. With no broad range gas detection system OPERABLE, within 1 hour initiate and maintain operation of the control room ventilation system in the isolate mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.3 Each broad range gas detection system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK ~~at least once per 12 hours~~, and a CHANNEL FUNCTIONAL TEST ~~at least once per 31 days~~. The CHANNEL FUNCTIONAL TEST will include the introduction of a standard gas.

Add INSERT 1a

*Including Ammonia

** The requirements of Technical Specification 3.0.1 do not apply during the time (two minutes or less) when the instrument automatic background/reference spectrum check renders the instrument(s) inoperable.

TABLE 4.3-9

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	~~D~~	N.A.	Q(4)	~~M~~	**
b. Oxygen Monitors	~~D~~	N.A.	~~Q~~(5)	~~M~~	**

Replace each marked through surveillance frequency with 'SFCP'.