Davis-Besse Nuclear Power Station
Robert B. Craven
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Oak Harbor, Ohio 43449

L-25-186 10 CFR 50.90
July 01, 2026

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject:
Davis-Besse Nuclear Power Station, Unit No.1
Docket No. 50-346, License No. NPF-3
LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT RISK INFORMED COMPLETION TIMES TSTF-505, REVISION 2, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES – RITSTF INITIATIVE 4B" AND TSTF-591, REVISION 0, "REVISE RISK INFORMED COMPLETION TIME (RICT) PROGRAM"

Pursuant to 10 CFR 50.90, Vistra Operations Company LLC (Vistra OpCo) is requesting an amendment to the Technical Specifications (TS) of the Davis-Besse Nuclear Power Station Unit 1 (DBNPS).

The proposed amendment would modify TS requirements to permit the use of Risk Informed Completion Times in accordance with TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b" (ADAMS Accession No. ML18183A493). A model safety evaluation with Limitations and Conditions was provided by the Nuclear Regulatory Commission (NRC) to the TS Task Force (TSTF) on November 21, 2018 (ADAMS Accession No. ML18267A259). The proposed amendment also incorporates NRC-approved traveler TSTF-591, "Revise the Risk Informed Completion Time (RICT) Program," (ADAMS Accession No. ML22081A224). A safety evaluation for TSTF-591 was provided by the NRC to the TSTF on September 21, 2023 (ADAMS Accession No. ML23262B230).

- Attachment 1 provides a description and assessment of the proposed change, the requested confirmation of applicability, and plant-specific verifications.
- Attachment 2 provides the existing TS pages marked up to show the proposed changes.
- Attachment 3 provides existing TS Bases pages marked up to show the proposed changes and is provided for information only.
- Attachment 4 provides a cross-reference between the TS included in TSTF-505, Revision 2, and the DBNPS plant-specific TS.
- Attachment 5 provides information supporting the redundant means available to mitigate accidents for instrumentation governed by the TS that are proposed to be included as part of the RICT program in this submittal.

Vistra OpCo requests approval of the proposed license amendment within 12 months following acceptance, with the amendment being implemented within 180 days of issuance of amendment.

In accordance with 10 CFR 50.91(a)(1), "Notice for Public Comment," the analysis about the issue of no significant hazards consideration using the standards in 10 CFR 50.92 is being provided to the Commission.

In accordance with 10 CFR 50.91(b)(1), "Notice for Public Comment; State Consultation," a copy of this application, with attachments, is being provided to the designated State of Ohio Official.

This letter and its attachments and enclosures contain no regulatory commitments.

If there are any questions, or if additional information is required, please contact Mr. Jack Hicks, Senior Manager, Licensing, at (254) 897-6725.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on July 1st, 2026.

Sincerely,

ALAN FILIPIAK FOR ROBERT CRAVEN VIA TELECOM 6/30/26

Robert B. Craven,
Site Vice President

Attachments:

1. Description and Assessment
2. Technical Specification Page Markups
3. Technical Specification Bases Page Markups – For Information Only
4. Cross-Reference of TSTF-505 and DBNPS Technical Specifications
5. Evaluation of Instrumentation and Control Systems

Enclosures:

1. List of Revised Required Actions to Corresponding PRA Functions
2. Information Supporting Consistency with Regulatory Guide 1.200, Revision 3
3. Information Supporting Technical Adequacy of PRA Models Without PRA Standards Endorsed by Regulatory Guide 1.200, Revision 3
4. Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models
5. Baseline CDF and LERF

6. Justification of Application of At-Power PRA Models to Shutdown Modes
7. PRA Model Update Process
8. Attributes of the Real-Time Model
9. Key Assumptions and Sources of Uncertainty
10. Program Implementation
11. Monitoring Program
12. Risk Management Action Examples

cc: NRC Region III Administrator
NRC Resident Inspector
NRR Project Manager
Utility Radiological Safety Board
Executive Director, Ohio Emergency Mgmt. Agency

Attachment 1

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Description and Assessment

TABLE OF CONTENTS

DESCRIPTION AND ASSESSMENT

1.0 SUMMARY DESCRIPTION

The proposed amendment would modify the Technical Specification (TS) requirements related to Completion Times (CTs) for Required Actions to provide the option to calculate a longer, risk-informed CT (RICT). A new program, the Risk-Informed Completion Time Program, and a new reporting requirement, the Risk-Informed Completion Time Program Upgrade Report, are added to TS Section 5.0, Administrative Controls.

The methodology for using the RICT Program is described in Nuclear Energy Institute (NEI) industry guidance document NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, which was approved by the Nuclear Regulatory Commission (NRC) on May 17, 2007 (Reference 1). Adherence to NEI 06-09-A is required by the RICT Program.

The proposed amendment is consistent with TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b" (Reference 2) as modified by TSTF-591, Revision 0, "Revise the Risk Informed Completion Time (RICT) Program". However, only those Required Actions described in Attachment 4 and Enclosure 1, as reflected in the proposed TS mark-ups provided in Attachments 2, are proposed to be changed. This is because some of the modified Required Actions in TSTF-505 are not applicable to Davis-Besse Nuclear Power Station Unit 1 (DBNPS).

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Vistra Operations Company LLC (Vistra OpCo) has reviewed TSTF-505, Revision 2, and the model safety evaluation dated November 21, 2018 (Reference 3). This review included the supporting information provided to support TSTF-505 and the safety evaluation for NEI 06-09-A. Vistra OpCo has also reviewed TSTF-591, Revision 0, and the model safety evaluation dated September 21, 2023 (ADAMS Accession No. ML23262B230). As described in the subsequent paragraphs, Vistra OpCo has concluded that the technical basis is applicable to DBNPS and supports incorporation of this amendment in the DBNPS TS.

2.2 Verifications and Regulatory Commitments

In accordance with Section 4.0, Limitations and Conditions, of the safety evaluation for NEI 06-09-A, the following is provided:

1. Enclosure 1 identifies each of the TS Required Actions to which the RICT Program will apply, with a comparison of the TS functions to the functions modeled in the probabilistic risk assessment (PRA) of the structures, systems and components (SSCs) subject to those actions.

2. Enclosure 2 provides a discussion of the results of peer reviews and self-assessments conducted for the plant-specific PRA models which support the RICT Program, as

discussed in Regulatory Guide (RG) 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3, Section 4.2.

3. Enclosure 3 is not applicable since each PRA model used for the RICT Program is addressed using a standard endorsed by the NRC.

4. Enclosure 4 provides appropriate justification for excluding sources of risk not addressed by the PRA models.

5. Enclosure 5 provides the plant-specific baseline core damage frequency (CDF) and large early release frequency (LERF) to confirm that the potential risk increases allowed under the RICT Program are acceptable.

6. Enclosure 6 is not applicable since the RICT Program is not being applied to shutdown modes.

7. Enclosure 7 provides a discussion of Vistra OpCo programs and procedures that will assure the PRA models that support the RICT Program are maintained consistent with the as-built, as-operated plant.

8. Enclosure 8 provides a description of how the baseline PRA model, which calculates average annual risk, is evaluated and modified to assess real-time configuration risk, and describes the scope of, and quality controls applied to the real-time model.

9. Enclosure 9 provides a discussion of how the key assumptions and sources of uncertainty in the PRA models were identified, and how their impact on the RICT Program was assessed and dispositioned.

10. Enclosure 10 provides a description of the implementing programs and procedures regarding the plant staff responsibilities for the RICT Program implementation, including risk management action (RMA) implementation.

11. Enclosure 11 provides a description of the implementation and monitoring program as described in NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, Section 2.3.2, Step 7.

12. Enclosure 12 provides a description of the process to identify and provide RMAs.

2.3 <u>Optional Variations</u>

Vistra OpCo is proposing the following variations from the TS changes described in TSTF-505, Revision 2, or the applicable parts of the NRC staff’s model safety evaluation dated November 21, 2018. These options were recognized as acceptable variations in TSTF-505 and the NRC staff's model safety evaluation. No variations from the TS changes described in TSTF-591, Revision 0, are proposed. The TSTF-591 justification states that adoption is dependent on previous adoption of TSTF-505, which is satisfied by adopting both travelers.

DBNPS is a Babcock and Wilcox Pressurized Water Reactor (PWR) design, and its Technical Specifications align with NUREG-1430, "Standard Technical Specifications, Babcock and

Wilcox Plants" (Reference 4). Attachment 4 is a cross-reference of the TSTF-505 Standard Technical Specification (STS) changes and the DBNPS Technical Specifications included in this license amendment request. Where the changes are identical, a disposition of “No variation” is provided. Where a variation is taken, the disposition provides a cross-reference to the paragraph below that provides justification.

2.3.1 Administrative Variations

The following variations taken from the TSTF-505 template for NUREG-1430 are considered to be administrative in nature.

2.3.1.1. DBNPS TS limiting conditions for operation (LCOs) and Required Actions (RAs) that differ from NUREG-1430 for numbering, titles, wording, and Completion Times while having a similar intent are administrative variations from TSTF-505 with no effect on the NRC staff's model safety evaluation.

2.3.1.2. For NUREG-1430 RAs that are not contained in the DBNPS TS, the corresponding NUREG-1430 markups included in TSTF-505 for these RAs and Completion Times are not applicable to DBNPS. These are administrative variations from TSTF-505 with no effect on the NRC staff's model safety evaluation.

2.3.1.3. Various TSTF-505 Section 3.3 instrumentation Conditions are invoked by instrumentation Functions contained in tables. The analogous DBNPS instrumentation Functions may have different Conditions referenced or have slightly different wording. This includes differences in lettering in the tables. These are administrative variations from TSTF-505 that meet the criteria for administering a RICT and have no effect on the NRC staff's model safety evaluation.

2.3.1.4. TSTF-505 applies RICT to certain RAs that require additional plant-specific justification. In any case where the DBNPS design has not been able to meet the necessary justification, a RICT has not been applied. For example, as the proposed DBNPS RICT Program is applicable in Modes 1 and 2, Vistra OpCo will not adopt changes in TSTF-505 for RAs that are only applicable in Mode 3 and below. Choosing to not apply a RICT under these circumstances are administrative variations and have no effect on the NRC staff’s model safety evaluation.

2.3.1.5. A RICT is not adopted for certain DBNPS TSs that may be contained in the TSTF-505 markups because it could not be verified that there would not be a loss of function, the function was not adequately modeled in the PRA, or an appropriate surrogate was not available. Choosing to not apply a RICT under these circumstances are administrative variations and have no effect on the NRC staff’s model safety evaluation.

2.3.1.6. For certain Conditions that could result in a loss of function, a note is added to the proposed statement "OR in accordance with the Risk Informed Completion Time

Program" to ensure that a RICT is not applied when the safety function is lost. A loss of function (LOF) is a loss of safety function as defined per the Safety Function Determination Program, in accordance with TS 5.5.14. Under this circumstance, TSTF-505, Revision 2, specifies the addition of a Note that reads "Not applicable when [all] required [channels] are inoperable." DBNPS has chosen to replace the TSTF-505 Note with a Note which reads "Not applicable when a loss of function occurs," which accomplishes the intended purpose of the TSTF-505 Note. These are administrative variations from TSTF-505 with no effect on the NRC staff's model safety evaluation.

2.3.1.7. The model application provided in TSTF-505, Revision 2 states the following in the markup for TS 5.5.18.e: “Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, …” The NRC staff’s model safety evaluation has alternate phrasing: “Methods to assess the risk from extending the Completion Times must be PRA methods approved for use with this program, …” Vistra OpCo has determined that the NRC phrasing is more appropriate, which is reflected in Attachment 2.

2.3.1.8. The model application provided in TSTF-505, Revision 2 includes an attachment for revised (clean) TS pages reflecting the proposed changes. DBNPS is not including such an attachment due to the number of TS pages included in this submittal that have the potential to be affected by other unrelated license amendment requests and the straightforward nature of the proposed changes. 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,” in that the mark-ups fully describe the changes desired. This is an administrative deviation from TSTF-505 with no impact on the NRC model safety evaluation.

2.3.2 Technical Variations

There are no variations from the TSTF-505 template for NUREG-1430 that are considered to be technical in nature.

2.4 Bases Changes

Revised TS Bases are provided in Attachment 3 for NRC information only. These Bases revisions will be incorporated as an implementing action pursuant to TS 5.5.13, Technical Specifications (TS) Bases Control Program, following issuance of the amendment.

3.0 REGULATORY SAFETY ANALYSIS

3.1 No Significant Hazards Consideration Analysis

Vistra Operations Company LLC (Vistra OpCo) has evaluated the proposed changes to the TS using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration.

Davis-Besse Nuclear Power Station Unit 1 (DBNPS), requests adoption of an approved change to the standard technical specifications (STS), to modify the TS requirements related to Completion Times for Required Actions to provide the option to calculate a longer, risk-informed Completion Time. The allowance is described in a new program in Chapter 5, "Administrative Controls," entitled the "Risk Informed Completion Time Program." A new report is added to Chapter 5 to inform the NRC of newly developed methods used to calculate a risk-informed Completion Time.

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

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes permit the extension of Completion Times provided the associated risk is assessed and managed in accordance with the NRC approved Risk-Informed Completion Time Program. The proposed changes do not involve a significant increase in the probability of an accident previously evaluated because the changes involve no change to the plant or its modes of operation. The proposed changes do not increase the consequences of an accident because the design-basis mitigation function of the affected systems is not changed and the consequences of an accident during the extended Completion Time are no different from those during the existing Completion Time.

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

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

   Response: No.

   The proposed changes do not change the design, configuration, or method of operation of the plant. The proposed changes do not involve a physical alteration of the plant (no new or different kind of equipment will be installed).

   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 a margin of safety?

   Response: No.

   The proposed changes permit the extension of Completion Times provided that risk is assessed and managed in accordance with the NRC approved Risk-Informed Completion Time Program. The proposed changes implement a risk-informed configuration management program to assure that adequate margins of safety are maintained. Application of these new specifications and the configuration management program considers cumulative effect of multiple systems or components being out of service and does so more effectively than the current TS.

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

Based on the above, Vistra OpCo concludes that the proposed changes present no significant hazards consideration under standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

3.2 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

Vistra OpCo has reviewed the environmental evaluations included in the model safety evaluation for TSTF-505, Revision 2, published on November 21, 2018 (ADAMS Accession No. ML18267A259) as part of the Notice of Availability and for TSTF-591, Revision 0, published on

December 18, 2023 (ADAMS Accession No. ML23325A214). Vistra OpCo has concluded that the NRC staff findings presented in those evaluations are applicable to DBNPS.

A review has determined the proposed change qualifies for a categorical exclusion from environmental review in accordance with 10 CFR 51.22. The proposed action is an amendment regarding the installation or use of a facility component as described in 10 CFR 51.22(d)(8) and involves changes to surveillance, inspection, or testing requirements as described in 10 CFR 51.22(d)(1). The proposed action does not disturb any previously undisturbed ground and there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, does not result in a significant increase in individual or cumulative public or occupational radiation exposure, and does not result in a significant increase in the potential for or consequences from radiological accidents.

Therefore, pursuant to 10 CFR 51.22, neither an environmental assessment nor an environmental impact statement is required.

5.0 REFERENCES

1. NRC letter to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines'," dated May 17, 2007 (ADAMS Accession No. ML071200238)

2. TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," (ADAMS Accession No. ML18183A493)

3. NRC letter to Technical Specifications Task Force, "Final Revised Model Safety Evaluation of Traveler TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b'," dated November 21, 2018 (ADAMS Accession No. ML18253A085)

4. NRC NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants," Revision 5, Volumes 1 and 2, dated September 2021

5. TSTF-591, Revision 0, "Revise Risk Informed Completion Time (RICT) Program," (ADAMS Accession No. ML22081A224)

Attachment 2

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Technical Specification Page Markups

INSERT EXAMPLE 1.3-8

EXAMPLE 1.3-8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
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

When a subsystem is declared inoperable, Condition A is entered. The 7 day Completion Time may be applied as discussed in Example 1.3-2. However, the licensee may elect to apply the Risk Informed Completion Time Program which permits calculation of a Risk Informed Completion Time (RICT) that may be used to complete the Required Action beyond the 7 day Completion Time. The RICT cannot exceed 30 days. After the 7 day Completion Time has expired, the subsystem must be restored to OPERABLE status within the RICT or Condition B must also be entered.

The Risk Informed Completion Time Program requires recalculation of the RICT to reflect changing plant conditions. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

If the 7 day Completion Time clock of Condition A has expired and subsequent changes in plant condition result in exiting the applicability of the Risk Informed Completion Time Program without restoring the inoperable subsystem to OPERABLE status, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start.

If the RICT expires or is recalculated to be less than the elapsed time since the Condition was entered and the inoperable subsystem has not been restored to OPERABLE status, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable subsystems are restored to OPERABLE status after Condition B is entered, Condition A is exited, and therefore, the Required Actions of Condition B may be terminated.

INSERT RICT 1

OR

In accordance with the Risk Informed Completion Time Program

INSERT RICT 2

OR

-------NOTE-------
Not applicable when a loss of function occurs.

In accordance with the Risk Informed Completion Time Program

INSERT RICT PROGRAM

5.5.20 Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODES 1 and 2;

c. When a RICT is being used, any change to the plant configuration, as defined in NEI 06-09-A, Appendix A, must be considered for the effect on the RICT.

1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. For emergent conditions, if the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the functions(s) performed by the inoperable SSCs.

e. A RICT calculation must include the following hazard groups: internal flood and internal events PRA model, internal fire PRA model, and seismic penalty factor. Changes to these means of assessing the hazard groups require prior NRC approval.

f. The PRA models used to calculate a RICT shall be maintained and upgraded in accordance with the processes endorsed in the regulatory positions of Regulatory Guide 1.200, Revision 3, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities."

g. A report shall be submitted in accordance with Specification 5.6.8 before a newly developed method is used to calculate a RICT.

INSERT RICT PROGRAM UPGRADE REPORT

5.6.8 Risk Informed Completion Time (RICT) Program Upgrade Report

A report describing newly developed methods and their implementation must be submitted following a probabilistic risk assessment (PRA) upgrade associated with newly developed methods and prior to the first use of those methods to calculate a RICT. The report shall include:

a. The PRA models upgraded to include newly developed methods;

b. A description of the acceptability of the newly developed methods consistent with Section 5.2 of PWROG-19027-NP, Revision 2, "Newly Developed Method Requirements and Peer Review;"

c. Any open findings from the peer-review of the implementation of the newly developed methods and how those findings were dispositioned; and

d. All changes to key assumptions related to newly developed methods or their implementation.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	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

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

INSERT EXAMPLE 1.3-8 →

3.3 INSTRUMENTATION

3.3.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1 Four channels of RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

AND

The ultrasonic flow meter (UFM) instrumentation shall be used to perform SR 3.3.1.2 when THERMAL POWER is > 50% RTP.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Place channel in bypass or trip.	1 hour
B. Two channels inoperable.	B.1 Place one channel in trip. AND B.2 Place second channel in bypass.	1 hour INSERT RICT 1 1 hour
C. Required Action and associated Completion Time of Condition A or B not met. OR Three or more channels inoperable.	C.1 Enter the Condition referenced in Table 3.3.1-1 for the Function.	Immediately

3.3 INSTRUMENTATION

3.3.5 Safety Features Actuation System (SFAS) Instrumentation

LCO 3.3.5 Four channels of SFAS instrumentation for each Parameter in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Parameters with one channel inoperable.	A.1	Place channel in trip.	1 hour INSERT RICT 1
B. Required Action and associated Completion Time of Condition A not met. OR One or more Parameters with two or more channels inoperable.	B.1 AND B.2	Be in MODE 3. -----NOTE----- Only required for Reactor Coolant System (RCS) Pressure - Low channels. ---------- Reduce RCS pressure < 1800 psig.	6 hours 36 hours
	AND		

3.3 INSTRUMENTATION

3.3.6 Safety Features Actuation System (SFAS) Manual Initiation

LCO 3.3.6 Two manual initiation channels of each one of the SFAS Functions below shall be OPERABLE:

a. SFAS; and

b. Containment Spray.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when associated engineered safety features equipment is required to be OPERABLE.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SFAS Functions with one channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours INSERT RICT 1
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 3.3.6.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.8 Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)

LCO 3.3.8 Two channels of Loss of Voltage Function and two channels of Degraded Voltage Function EDG LOPS instrumentation per bus shall be OPERABLE.

-----NOTE-----
All Degraded Voltage channels may be bypassed for ≤ 1 minute when starting a reactor coolant pump or circulating water pump.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated EDG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel per bus inoperable.	A.1 Place channel in trip.	1 hour INSERT RICT 2
B. One or more Functions with two channels per bus inoperable.	B.1 Restore one channel to OPERABLE status.	1 hour INSERT RICT 2
C. Required Action and associated Completion Time not met.	C.1 Declare associated EDG inoperable.	Immediately

3.3 INSTRUMENTATION

3.3.12 Steam and Feedwater Rupture Control System (SFRCS) Manual Initiation

LCO 3.3.12 One manual initiation push button for each of the following SFRCS Functions shall be OPERABLE:

a. Auxiliary Feedwater Pump Turbine 1 Initiation;

b. Auxiliary Feedwater Pump Turbine 2 Initiation;

c. Auxiliary Feedwater Pump Turbine 1 Initiation and Steam Generator 1 Isolation; and

d. Auxiliary Feedwater Pump Turbine 2 Initiation and Steam Generator 2 Isolation.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SFRCS Functions inoperable.	A.1 Restore SFRCS Function to OPERABLE status.	48 hours INSERT RICT 1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

3.3 INSTRUMENTATION

3.3.13 Steam and Feedwater Rupture Control System (SFRCS) Actuation

LCO 3.3.13 Channels 1 and 2 of each Logic Function shown below shall be OPERABLE:

a. Auxiliary Feedwater Initiation;

b. Auxiliary Feedwater and Main Steam Valve Control;

c. Main Steam Line Isolation; and

d. Main Feedwater Isolation.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Logic Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channel 1 Logic Functions inoperable with all channel 2 Logic Functions OPERABLE. OR One or more channel 2 Logic Functions inoperable with all channel 1 Logic Functions OPERABLE.	A.1 Restore inoperable channel to OPERABLE status.	72 hours INSERT RICT 1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 228 inches; and

b. A minimum of 85 kW of essential pressurizer heaters OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Restore level to within limit.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours
C. Capacity of essential pressurizer heaters less than limit.	C.1 Restore essential pressurizer heater capacity.	72 hours INSERT RICT 1
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

-----NOTE-----
The borated water storage tank (BWST) outlet and containment emergency sump valves may be considered OPERABLE when the associated valve motors are de-energized, provided the valves are not otherwise inoperable.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure injection (LPI) subsystem inoperable.	A.1 Restore LPI subsystem to OPERABLE status.	7 days INSERT RICT 1
B. One or more trains inoperable for reasons other than Condition A.	B.1 Restore train(s) to OPERABLE status.	72 hours INSERT RICT 1
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours
D. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	D.1 Enter LCO 3.0.3.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Lock an OPERABLE door closed in the affected air lock.	24 hours
	AND		
	B.3	-----------NOTE----------- Air lock doors in high radiation areas may be verified locked closed by administrative means. --------------------------------	
		Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	AND		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours INSERT RICT 1
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

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

ACTIONS

-----NOTES-----

1. Penetration flow paths except for 48 inch containment purge and exhaust valve penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for system(s) made inoperable by containment isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. -----NOTE----- Only applicable to penetration flow paths with two or more containment isolation valves. ----- One or more penetration flow paths with one containment isolation valve inoperable for reasons other than Condition D or E.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours INSERT RICT 1
	AND		

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	-----------NOTES----------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ----------------------------------- Verify the affected penetration flow path is isolated.	following isolation Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
B. -----------NOTE----------- Only applicable to penetration flow paths with two or more containment isolation valves. ----------------------------------- One or more penetration flow paths with two or more containment isolation valves inoperable for reasons other than Condition D or E.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE---------- Only applicable to penetration flow paths with only one containment isolation valve. -------------------------------- One or more penetration flow paths with containment isolation valve inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours INSERT RICT 1
	AND	
	C.2 ----------NOTES---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. -------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days following isolation

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Air Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment air cooling trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	7 days INSERT RICT 1
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours
C. One required containment air cooling train inoperable.	C.1 Restore required containment air cooling train to OPERABLE status.	7 days INSERT RICT 1
D. One containment spray train and one required containment air cooling train inoperable.	D.1 Restore containment spray train to OPERABLE status. OR D.2 Restore required containment air cooling train to OPERABLE status.	72 hours INSERT RICT 1 72 hours INSERT RICT 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two required containment air cooling trains inoperable.	E.1 Restore one required containment air cooling train to OPERABLE status.	72 hours INSERT RICT 1
F. Required Action and associated Completion Time of Condition C, D, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours
G. Two containment spray trains inoperable. OR Any combination of three or more required trains inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray 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.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.2	Operate each required containment air cooling train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MSIVs are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	8 hours INSERT RICT 1
B. More than one MSIV inoperable in MODE 1. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
C. ------------NOTE------------ Separate Condition entry is allowed for each MSIV. -------------------------------- One or more MSIVs inoperable in MODE 2 or 3.	C.1 Close MSIV. <u>AND</u> C.2 Verify MSIV is closed.	8 hours Once per 7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW)

LCO 3.7.5 Three EFW trains shall be OPERABLE, consisting of:

a. Two Auxiliary Feedwater (AFW) trains; and

b. The Motor Driven Feedwater Pump (MDFP) train.

-----NOTE-----
Only the MDFP train is required to be OPERABLE in MODE 4.

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

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable when entering MODE 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AFW train inoperable due to one inoperable steam supply. OR -----NOTE----- Only applicable if MODE 2 has not been entered following refueling. --- One AFW train inoperable in MODE 3 following refueling.	A.1 Restore AFW train to OPERABLE status.	7 days INSERT RICT 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One EFW train inoperable for reasons other than Condition A in MODE 1, 2, or 3.	B.1	Restore EFW train to OPERABLE status.	72 hours INSERT RICT 1
C. One AFW train inoperable due to one inoperable steam supply. AND MDFP train inoperable.	C.1	Restore the steam supply to the AFW train to OPERABLE status.	48 hours
	OR		
	C.2	Restore the MDFP train to OPERABLE status.	48 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met. OR Two EFW trains inoperable for reasons other than Condition C in MODE 1, 2, or 3.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4.	12 hours
E. Three EFW trains inoperable in MODE 1, 2, or 3.	E.1	-----------NOTE----------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. -------------------------------- Initiate action to restore one EFW train to OPERABLE status.	Immediately

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW loops shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW loop inoperable.	A.1	------------NOTES------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by CCW. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for decay heat removal loop made inoperable by CCW. --------------------------------- Restore CCW loop to OPERABLE status.	72 hours INSERT RICT 1
B. Required Action and associated Completion Time not met.	B.1 <u>AND</u> B.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 Two SWS loops shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SWS loop inoperable.	A.1	-----------NOTES----------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by SWS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for decay heat removal loop made inoperable by SWS. -------------------------------- Restore SWS loop to OPERABLE status.	72 hours INSERT RICT 1
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore offsite circuit to OPERABLE status.	72 hours* INSERT RICT 1
B. One EDG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s) supported by the inoperable EDG inoperable when its redundant required feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE EDG is not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE EDG.	24 hours
	AND		
	B.4	Restore EDG to OPERABLE status.	7 days INSERT RICT 1

* A one-time Completion Time (CT) extension for an inoperable offsite circuit allows 15 days to restore the inoperable Startup Transformer X02 to OPERABLE status. Compensatory measures within Vistra Operations Company LLC letter L-25-227 dated 12/22/2025 shall be implemented and shall remain in effect during the extended CT period. The one-time extension shall expire upon completion of the maintenance to restore Startup Transformer X02 to OPERABLE status or by 0600 on March 6, 2026, whichever occurs earliest.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND	
	C.2 Restore one offsite circuit to OPERABLE status.	24 hours INSERT RICT 1
D. One offsite circuit inoperable. AND One EDG inoperable.	-----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating," when Condition D is entered with no AC power source to any train. -----	
	D.1 Restore offsite circuit to OPERABLE status.	12 hours INSERT RICT 1
	OR	
	D.2 Restore EDG to OPERABLE status.	12 hours INSERT RICT 1
E. Two EDGs inoperable.	E.1 Restore one EDG to OPERABLE status.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 5.	6 hours 36 hours
G. ----------NOTE---------- Separate Condition entry is allowed for each train. ------------------------------- One or more trains with one load sequencer inoperable.	G.1 Remove inoperable load sequencer.	1 hour INSERT RICT 1
H. ----------NOTE---------- Separate Condition entry is allowed for each train. ------------------------------- Required Action and associated Completion Time of Condition G not met. OR One or more trains with two load sequencers inoperable.	H.1 Declare associated EDG inoperable.	Immediately
I. Three or more AC sources inoperable.	I.1 Enter LCO 3.0.3.	Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 The Train 1 and Train 2 DC electrical power sources shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required battery chargers on one train inoperable.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage. <u>AND</u> A.2 Verify battery float current ≤ 2 amps. <u>AND</u> A.3 Restore required battery charger(s) to OPERABLE status.	2 hours Once per 12 hours 72 hours INSERT RICT 1
B. One DC electrical power source inoperable for reasons other than Condition A.	B.1 Restore DC electrical power source to OPERABLE status.	2 hours INSERT RICT 1
C. Two DC electrical power sources inoperable. <u>OR</u> 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

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

LCO 3.8.7 The Train 1 and Train 2 inverters shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One inverter inoperable.	A.1	-------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any 120 VAC vital bus de-energized. ------------------------------------- Restore inverter to OPERABLE status.	24 hours INSERT RICT 1
B. Two inverters in one train inoperable.	B.1	Restore one inverter to OPERABLE status	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 Train 1 and Train 2 AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystems inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," for DC sources made inoperable by inoperable power distribution subsystems. ------------------------------	
	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours INSERT RICT 1
B. One or more AC vital buses inoperable.	B.1 Restore AC vital bus(es) to OPERABLE status.	8 hours INSERT RICT 1
C. One DC electrical power distribution subsystems inoperable.	C.1 Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours INSERT RICT 1

5.5 Programs and Manuals

5.5.19 Online Monitoring Program (continued)

b. Performance of a calibration check for any transmitter where the online monitoring was not implemented during the plant operating cycle no later than during the next refueling outage.

c. Performance of calibration checks for transmitters at the specified backstop frequencies.

d. The provisions of Surveillance Requirement 3.0.3 are applicable to the required calibration checks specified in items a.3, b, and c above.

INSERT RICT PROGRAM

5.6 Reporting Requirements

5.6.6 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG;

b. Degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;

e. Number of tubes plugged during the inspection outage for each degradation mechanism;

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG;

g. The results of condition monitoring, including the results of tube pulls and in-situ testing;

5.6.7 Remote Shutdown System Report

When a report is required by Condition C of LCO 3.3.18, "Remote Shutdown System," a report shall be submitted within the following 30 days. The report shall outline the action taken, the cause of the inoperability, and the plans and schedule for restoring the control circuit or transfer switch of the Function to OPERABLE status.

INSERT RICT PROGRAM UPGRADE REPORT

Attachment 3

L-25-186

Technical Specifications Bases Page Markups
For Information Only

INSERT 1

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

INSERT 2

or in accordance with the Risk Informed Completion Time Program

INSERT 3

The Completion Time using the Risk Informed Completion Time Program is modified by a Note to clarify that it is not applicable when a loss of safety function occurs as defined in TS 5.5.14. A loss of safety function would exist if one or more loss of voltage channels on both 4.16 kV essential buses are failed OR one or more degraded voltage channels on both 4.16 kV essential buses are failed. Either of these configurations would result in defeating automatic EDG LOPS signals when required. Therefore, a RICT cannot be applied when such a configuration exists.

BASES

ACTIONS

Conditions A, B, and C are applicable to all RPS protection Functions. If a channel's trip setpoint is found non-conservative with respect to the required Allowable Value in Table 3.3.1-1, or the transmitter, instrument loop, signal processing electronics or bistable is found inoperable, the channel must be declared inoperable and the applicable Condition must be entered immediately.

A.1

If one or more Functions in one protection channel become inoperable, the affected protection channel must be placed in bypass or trip. If the channel is bypassed, all RPS Functions are placed in a two-out-of-three logic configuration and the bypass of any other channel is prevented. In this configuration, the RPS can still perform its safety function in the presence of a random failure of any single channel. Alternatively, the inoperable channel can be placed in trip. Tripping the affected protection channel places all RPS Functions in a one-out-of-three configuration.

Operation in the two-out-of-three configuration or in the one-out-of-three configuration may continue indefinitely based on the NRC SER for BAW-10167, Supplement 2 (Ref. 9). In this configuration, the RPS is capable of performing its trip Function in the presence of any single random failure. The 1 hour Completion Time is sufficient to perform Required Action A.1.

B.1 and B.2

For Required Action B.1 and Required Action B.2, if one or more Functions in two protection channels become inoperable, one of two inoperable protection channels must be placed in trip and the other in bypass. These Required Actions place all RPS Functions in a one-out-of-two logic configuration and prevent bypass of a second channel. In this configuration, the RPS can still perform its safety functions in the presence of a random failure of any single channel. The 1 hour Completion Time is sufficient time to perform Required Action B.1 and Required Action B.2.

Alternatively, a Completion Time can be determined for Required Action B.1 in accordance with the Risk Informed Completion Time Program.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1. The applicable Condition referenced in the table is Function dependent. If the Required Action and the associated Completion Time of Condition A or B are not met or if more than two channels are inoperable, Condition C is entered to provide for transfer to the appropriate subsequent Condition.

BASES

ACTIONS

Required Actions A and B apply to all SFAS instrumentation Parameters listed in Table 3.3.5-1.

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each Parameter.

If a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or SFAS bistable is found inoperable, then all affected functions provided by that channel should be declared inoperable and the unit must enter the Conditions for the particular protection Parameter affected.

A.1

Condition A applies when one channel becomes inoperable in one or more Parameters. If one SFAS channel is inoperable, placing it in a tripped condition leaves the system in a one-out-of-three condition for actuation. Thus, if another channel were to fail, the SFAS instrumentation could still perform its actuation functions. This action is completed when all of the affected logic inputs are tripped. This can normally be accomplished by tripping the affected bistables.

The 1 hour Completion Time is sufficient time to perform the Required Action.

INSERT 1

B.1, B.2.1, B.2.2, and B.2.3

Condition B applies when the Required Action and associated Completion Time of Condition A is not met or when one or more parameters have more than one inoperable channel. If Condition B applies, the unit must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and, for the RCS Pressure - Low channels, to < 1800 psig, for the RCS Pressure - Low Low channels, to < 660 psig, and for the Containment Pressure High channels and High High channels, to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

APPLICABLE SAFETY ANALYSES

The SFAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents, specifically, the loss of coolant accident and main steam line break events.

The SFAS manual initiation ensures that the control room operator can rapidly initiate ESF Functions at any time. The manual initiation trip Function is required as a backup to automatic trip functions and allows operators to initiate SFAS whenever any parameter is rapidly trending toward its trip setpoint. Furthermore, the SFAS manual initiation may be specified in operating procedures for verification of ESF actuation.

The SFAS manual initiation functions satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two SFAS manual initiation channels of each SFAS Function shall be OPERABLE whenever conditions exist that could require ESF protection of the reactor or containment. Two OPERABLE channels of SFAS manual initiation and two ~~OPERBLE~~ OPERABLE channels for Containment Spray manual initiation ensure that no single random failure will prevent system level manual initiation of any required SFAS Function. The SFAS manual initiation Function allows the operator to initiate protective action prior to automatic initiation or in the event the automatic initiation does not occur.

APPLICABILITY

The SFAS manual initiation Functions shall be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the associated engineered safety features equipment is required to be OPERABLE. The manual initiation channels are required because ESF Functions are designed to provide protection in these MODES. In MODES 5 and 6, SFAS initiates systems that are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and to respond by manually operating the ESF components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each SFAS manual initiation Function.

A.1

Condition A applies when one manual initiation channel of one or more SFAS Functions becomes inoperable. Required Action A.1 must be taken to restore the channel to OPERABLE status within the next 72 hours. [INSERT 2] The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means

BASES

ACTIONS (continued)

entered for the particular protection function affected. Since the required channels are specified on a per bus basis, the Condition may be entered separately for each bus.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function.

A.1

INSERT 2

If one channel per bus in one or more Functions is inoperable, it must be tripped within 1 hour. With a channel in trip, the EDG LOPS channels are configured to provide a one-out-of-one logic to generate an EDG LOPS signal. In trip, one additional valid actuation will cause an EDG LOPS signal on the bus. The 1 hour Completion Time is reasonable to evaluate and to take action by correcting a degraded condition in an orderly manner and takes into account the low probability of an event requiring an EDG LOPS occurring during this interval.

INSERT 3

B.1

Condition B applies when two undervoltage or two degraded voltage channels per bus are inoperable.

INSERT 2

Required Action B.1 requires one inoperable channel to be restored to OPERABLE status within 1 hour. With two channels as described above inoperable, the logic is not capable of providing an automatic EDG LOPS signal for valid loss of voltage or degraded voltage conditions. The 1 hour Completion Time is reasonable to evaluate and to take action by correcting the degraded condition in an orderly manner and takes into account the low probability of an event requiring an EDG LOPS occurring during this interval.

INSERT 3

C.1

Condition C applies if the Required Action of Condition A or Condition B and the associated Completion Time is not met.

Required Action C.1 requires the associated EDG to be declared inoperable, which ensures that Required Actions for affected EDG inoperabilities are initiated. Depending on unit MODE, the Actions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, are required immediately.

BASES

BACKGROUND (continued)

These Functions are provided in the event the operator determines that an SFRCS Function is needed and does not automatically actuate. These are backup Functions to those performed automatically by SFRCS.

The SFRCS manual initiation circuitry satisfies the manual initiation and single-failure criterion requirements of IEEE-279-1971 (Ref. 1).

APPLICABLE SAFETY ANALYSES

SFRCS Functions credited in the safety analysis are automatic. However, the manual initiation Functions are required by design as backups to the automatic trip Functions and allow operators to initiate AFPT and actuate SG isolation whenever these Functions are needed. Furthermore, the manual initiation of the AFPT and isolation of the SG may be specified in unit operating procedures.

The SFRCS manual initiation Functions are retained for the overall redundancy and diversity of the SFRCS as required by the NRC.

LCO

Each push button performing an SFRCS manual initiation Function shall be OPERABLE. Failure of any push button renders the affected Function inoperable.

One manual initiation push button for each Function (AFPT 1 Initiation (with the exception of ARTS and Main Turbine Trip); AFPT 2 Initiation (with the exception of ARTS and Main Turbine Trip); AFPT 1 Initiation and SG 1 Isolation; and AFPT 2 Initiation and SG 2 Isolation) is required to be OPERABLE whenever the SGs are being relied on to remove heat.

APPLICABILITY

The SFRCS manual initiation Functions shall be OPERABLE in MODES 1, 2, and 3 because the SGs are relied on for Reactor Coolant System heat removal and because SG inventory can be at a sufficiently high energy level to contribute significantly to the peak containment pressure during a secondary pipe break. In MODES 4, 5, and 6, heat removal requirements are reduced and can be provided by the Decay Heat Removal System because the SG energy level is low and secondary side feedwater flow rate is low or nonexistent.

ACTIONS

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each SFRCS manual initiation Function.

A.1

INSERT 2

With one or more SFRCS Functions inoperable the Function must be restored to OPERABLE status within 48 hours. The Completion Time

BASES

ACTIONS (continued)

A.1

Condition A applies when one or more SFRCS Logic Functions in a single channel are inoperable (i.e., channel 1 could be inoperable for all four SFRCS Logic Functions and Condition A would still be applicable) with all Logic Functions in the other channel OPERABLE. This Condition is equivalent to failure of one AFW, Main Steam Line Isolation, and MFW Isolation train.

With one channel of one or more SFRCS Logic Functions inoperable, the associated SFRCS train must be restored to OPERABLE status. Since there are only two automatic actuation logic channels per SFRCS Function, the condition of one channel inoperable is analogous to having one train of a two train Engineered Safety Feature (ESF) System inoperable. The system safety function can be accomplished; however, a single failure cannot be taken. Therefore, the failed channel(s) must be restored to OPERABLE status to re-establish the system's single-failure tolerance.

Condition A can be thought of as equivalent to failure of a single train of a two train safety system (e.g., the safety function can be accomplished, but a single failure cannot be taken). Thus, the Completion Time of 72 hours has been chosen to be consistent with Completion Times for restoring one inoperable ESF System train. ← INSERT 1

The SFRCS has not been analyzed for failure of Logic Functions in both channels. In this condition, the potential for system interactions that disable heat removal capability on AFW has not been evaluated. Consequently, any combination of failures in both channels 1 and 2 is not covered by Condition A and must be addressed by entry into LCO 3.0.3.

B.1 and B.2

If Required Action A.1 cannot be met within the associated Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least 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 MODES from full power conditions in an orderly manner and without challenging unit systems.

BASES

APPLICABILITY (continued)

MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbations, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Decay Heat Removal System is in service, and therefore the LCO is not applicable.

ACTIONS

A.1

With pressurizer water level in excess of the maximum limit, action must be taken to restore pressurizer operation to within the bounds assumed in the analysis. This is done by restoring the pressurizer water level to within the limit. The 1 hour Completion Time is considered to be a reasonable time for draining excess liquid.

B.1 and B.2

If the water level cannot be restored, reducing core power constrains heat input effects that drive pressurizer insurge that could result from an anticipated transient. Therefore, 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 to MODE 4 within 12 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable based on operating experience to achieve power reduction from full power conditions in an orderly manner and without challenging plant systems.

C.1

INSERT 2

If the essential pressurizer heaters are inoperable, restoration is required in 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power will not occur in this period. Pressure control may be maintained during this time using normal station powered heaters.

BASES

ACTIONS

A.1

INSERT 2

With one LPI subsystem inoperable, action must be taken to restore it to OPERABLE status within 7 days. In this condition, the remaining OPERABLE ECCS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure to the remaining LPI subsystem could result in loss of ECCS function. The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable LPI subsystem. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 3. Reference 3 concluded that extending the Completion Time to 7 days for an inoperable LPI subsystem improves plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the LPI subsystem unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations.

B.1

INSERT 2

With one or more trains inoperable and at least 100% of the injection flow equivalent to a single OPERABLE ECCS train available, components inoperable for reasons other than Condition A must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 4) that are based on a risk evaluation and is a reasonable time for many repairs.

An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 4) has shown the risk of having one full ECCS train inoperable to be sufficiently low to justify continued operation for 72 hours.

BASES

ACTIONS (continued)

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

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

INSERT 2

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the

BASES

ACTIONS (continued)

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for containment purge and exhaust valve leakage or secondary containment bypass leakage paths not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within the 4 hour Completion Time. INSERT 2 The specified time period is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices

BASES

ACTIONS

A.1 and A.2 (continued)

following isolation

outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating this Condition is only applicable to those penetration flow paths with two or more containment isolation valves. For penetration flow paths with only one containment isolation valve, Condition C provides appropriate actions.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows the devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

B.1

With two or more containment isolation valves in one or more penetration flow paths inoperable, except for containment purge and exhaust valve leakage or secondary containment bypass leakage paths not within limit, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic

BASES

ACTIONS

B.1 (continued)

verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two or more containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the 72 hour Completion Time. INSERT 2 The specified time period is reasonable, considering: the relative stability of the system (hence, reliability) to act as a penetration isolation boundary, since it does not communicate with the containment atmosphere or reactor coolant pressure boundary (for certain valves in Type III penetrations); the small pipe diameter of the penetration (hence, reliability) (for certain valves in Type II, III and IV penetrations); that the valves isolate Engineered Safety Features Systems that normally operate following an accident (for most valves in Type IV penetrations); and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days following isolation for verifying that each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

BASES

APPLICABILITY (continued)

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and the Containment Air Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

INSERT 2

With one containment spray train inoperable, action must be taken to restore it to OPERABLE status within 7 days. In this condition, the remaining OPERABLE containment spray train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure to the remaining containment spray train could result in loss of spray function. The 7 day Completion Time is reasonable to perform corrective maintenance on the inoperable containment spray train. The 7 day Completion Time is based on the findings of the deterministic and probabilistic analysis in Reference 5. Reference 5 concluded that extending the Completion Time to 7 days for an inoperable containment spray train improves plant operational flexibility while simultaneously reducing overall plant risk. This is because the risks incurred by having the containment spray train unavailable for a longer time at power will be substantially offset by the benefits associated with avoiding unnecessary plant transitions and by reducing risk during plant shutdown operations.

B.1 and B.2

If the inoperable containment spray train 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 84 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. The extended interval to reach MODE 5 allows additional time to attempt restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

BASES

ACTIONS (continued)

C.1

INSERT 2

With one of the required containment air cooling trains inoperable, the inoperable containment air cooling train must be restored to OPERABLE status within 7 days. The remaining OPERABLE containment spray and air cooling trains in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Air Cooling System and the low probability of a DBA occurring during this period.

D.1 and D.2

INSERT 2

With one containment spray and one required containment air cooling train inoperable, either one containment spray train or one of the required containment air cooling trains must be restored to OPERABLE status within 72 hours. The remaining OPERABLE containment spray and air cooling trains in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Air Cooling System, the iodine removal function of the Containment Spray System, and the low probability of a DBA occurring during this period.

E.1

INSERT 2

With two of the required containment air cooling trains inoperable, one of the required containment air cooling trains must be restored to OPERABLE status within 72 hours. The remaining OPERABLE containment spray trains in this degraded condition (both containment spray trains are OPERABLE or else Condition G is entered) provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Air Cooling System and the low probability of a DBA occurring during this period.

BASES

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1 and in MODES 2 and 3 with any MSIV open, when there is significant mass and energy in the RCS and steam generator; therefore, the MSIVs must be OPERABLE or closed. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low. Therefore, the MSIVs are not required to be OPERABLE.

In MODES 5 and 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

INSERT 2

With one MSIV inoperable in MODE 1, action must be taken to restore the MSIV to OPERABLE status within 8 hours. Some repairs can be made to the MSIV with the unit hot. The 8 hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the MSIVs occurring during this time interval. The turbine stop valves are also available to provide the required isolation for some accidents.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a penetration that is neither part of the reactor coolant pressure boundary nor is connected directly to the containment atmosphere.

B.1

in accordance with Required Action A.1

If the MSIV cannot be restored to OPERABLE status ~~within 8 hours~~, the unit must be placed in MODE 2 within the next 6 hours. The Completion Time is reasonable, based on operating experience, to reach MODE 2.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

BASES

APPLICABILITY

In MODES 1, 2, and 3, EFW is required to be OPERABLE and to function in the event that the main feedwater is lost. In addition, EFW is required to supply enough makeup water to replace the steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

In MODE 4, EFW may be used for heat removal via the steam generators. In MODE 4, the steam generators are used for heat removal until the DHR System is in operation.

In MODES 5 and 6, the steam generators are not used for heat removal and EFW is not required.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable EFW train when entering MODE 1. There is an increased risk associated with entering MODE 1 with EFW inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

INSERT 2

With one AFW train inoperable due to one inoperable steam supply, or if an AFW train is inoperable for any reason while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. For the inoperability of an AFW train due to one inoperable steam supply, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the AFW train and the AFW train is still capable of performing its specified function for most postulated events.

b. For the inoperability of an AFW train while in MODE 3 immediately subsequent to a refueling, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.

c. In addition, for either the inoperability of an AFW train due to one inoperable steam supply or an inoperable AFW train while in MODE 3 immediately following refueling, the 7 day Completion Time is reasonable due to the availability of redundant OPERABLE EFW pumps, and due to the low probability of an event requiring the use of the inoperable AFW pump.

Condition A is modified by a Note which limits the applicability of the Condition for an inoperable AFW pump in MODE 3 to when the unit has

BASES

ACTIONS

A.1 (continued)

not entered MODE 2 following a refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

B.1

INSERT 2

When one of the EFW trains (pump or flow path) is inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore the train to OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to one AFW train. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by EFW, time needed for repairs, and the low probability of a DBA occurring during this time period.

C.1 and C.2

With the MDFP train (pump or flow path) inoperable and one of the AFW trains inoperable due to one inoperable steam supply, action must be taken to restore the affected equipment to OPERABLE status within 48 hours. Assuming no single active failures when in this condition, the accident (a MSLB) could result in the loss of the remaining steam supply to the inoperable AFW pump due to the faulted steam generator. In this condition, the EFW may no longer be able to meet the required flow to the steam generators assumed in the safety analysis.

The 48 hour Completion Time is reasonable based on the fact that the remaining AFW train is capable of providing 100% of the EFW flow requirements, and the low probability of an event occurring that would challenge the EFW.

D.1 and D.2

When Required Action A.1, B.1, C.1, or C.2 cannot be completed within the required Completion Time, or when two EFW trains are inoperable for reasons other than Condition C in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," and LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable CCW loop results in an inoperable EDG or DHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

INSERT 2

If one CCW loop is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this condition, the remaining OPERABLE CCW loop is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE loop, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CCW loop cannot be restored to OPERABLE status in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

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 System.

Verifying the correct alignment for manual, power operated, and automatic 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 they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves which cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SWS, in conjunction with the CCW System, also cools the unit from Decay Heat Removal (DHR) System entry conditions to MODE 5 during normal and post accident operation, as discussed in the UFSAR, Section 6.3 (Ref. 3). The time required for this evolution is a function of the number of CCW and DHR System loops that are operating. One SWS train is sufficient to remove decay heat during subsequent operations with T_{cold} < 200°F. This assumes an initial SWS temperature of 90°F occurring simultaneously with maximum heat loads on the system.

The SWS is also required when needed to support CCW in the removal of heat from the emergency diesel generators (EDGs) or reactor auxiliaries.

The SWS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two SWS loops are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power.

An SWS loop is considered OPERABLE when:

a. It has an OPERABLE pump; and

b. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

Furthermore, the spare SWS pump can be substituted for a normal SWS pump, provided the power supply for the pump is aligned to the same essential bus as the pump it is replacing.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SWS and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the SWS are determined by the systems it supports.

ACTIONS

A.1

INSERT 2

If one SWS loop is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this condition, the remaining OPERABLE SWS loop is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SWS loop could result in loss of SWS function. Required Action A.1 is modified by two Notes. The first Note indicates that the

BASES

ACTIONS (continued)

A.2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated EDG will not result in a complete loss of safety function of critical redundant required features. These features are powered from the redundant AC electrical power train.

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The train has no offsite power supplying its loads; and

b. A redundant required feature on the other train is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable redundant required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Time for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to essential bus C1 and essential bus D1 of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

INSERT 1

Consistent with Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition,

BASES

ACTIONS (continued)

B.4 (continued)

System. The 7 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. The 7 day Completion Time is also acceptable as described in Reference 8. INSERT 1

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required features. These features are powered from redundant AC safety trains. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A redundant required feature is inoperable.

If at any time during the existence of Condition C (two offsite circuits inoperable) and a redundant required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours. INSERT 1 This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

BASES

ACTIONS (continued)

D.1 and D.2 (continued)

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 12 hours. INSERT 1

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period. INSERT 1

E.1

With both EDGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Regulatory Guide 1.93 (Ref. 6), with both EDGs inoperable, operation may continue for a period that should not exceed 2 hours.

F.1 and F.2

If the inoperable AC electrical power sources cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from

BASES

ACTIONS (continued)

F.1 and F.2 (continued)

full power conditions in an orderly manner and without challenging plant systems.

G.1

One of the SFAS actions during an Incident Level 2 is to start the EDG. In the event of a loss of offsite power concurrent with an SFAS trip, the SFAS sequencer will apply emergency loads to the essential bus in accordance with the sequencer load program. Each SFAS actuation channel has two load sequencer modules.

INSERT 1

With one or more trains with one load sequencer inoperable, the 1 hour Completion Time provides a period of time to remove the inoperable module from the SFAS cabinet. As noted, since each train is independent from the other train, separate Condition entry is allowed for inoperable load sequencers in each train.

H.1

With one or more trains with two load sequencers inoperable, the EDG cannot be loaded in the proper sequence and therefore, cannot meet its safety function. Therefore, the EDG must be immediately declared inoperable. As noted, since each train is independent from the other train, separate Condition entry is allowed for inoperable load sequencers in each train.

I.1

Condition I corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

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, in accordance with UFSAR, Section 8 (Ref. 9). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the EDGs are consistent with the

BASES

ACTIONS

A.1, A.2, and A.3 (conrinued)

amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 12 hours (Required Action A.2).

Required Action A.2 requires that the battery float current be verified as less than or equal to 2 amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial 12 hour period the battery float current is not less than or equal to 2 amps this indicates there may be additional battery problems and the battery must be declared inoperable.

INSERT 1

Required Action A.3 limits the restoration time for the inoperable required battery charger to 72 hours. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., balance of plant non-Class 1E battery charger, use of an inoperable, but functional Class 1E battery charger). The 72 hour Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

B.1

Condition B represents one train with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected train. The 2 hour limit is consistent with the allowed time for an inoperable DC distribution system train. INSERT 1

If one of the required DC electrical power sources is inoperable for reasons other than Condition A (e.g., one or both batteries in a train inoperable), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure could, however, result in the loss of minimum necessary DC electrical sources to mitigate a worst case accident, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 7) and reflects a reasonable time to assess unit status as a function

BASES

ACTIONS

A.1 (continued)

INSERT 1

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the 120 VAC vital bus is powered from its Class 1E constant voltage transformer, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the 120 VAC vital buses is the preferred source for powering instrumentation trip setpoint devices.

B.1

With two inverters in the same train inoperable, the remaining inverters are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in one of the two remaining inverters could result in the minimum ESF functions not being supported. Therefore, one of the inverters must be restored to OPERABLE status within 8 hours.

The 8 hour Completion Time is consistent with that allowed for an inoperable train of 120 VAC vital buses.

C.1 and C.2

If the inoperable Train 1 or Train 2 inverters cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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 unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and 120 VAC vital buses energized from the associated inverter. Each inverter may be connected to its associated rectifier as long as the battery is available as the

BASES

APPLICABILITY (continued)

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 5 and 6 and other conditions in which electrical power distribution subsystems are required are covered in the Bases for LCO 3.8.10, "Distribution Systems - Shutdown."

ACTIONS

A.1

With one or more Train 1 and Train 2 required AC electrical power distribution subsystems (except AC vital buses), inoperable and a loss of function has not occurred, the remaining AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC electrical power distribution subsystem(s) must be restored to OPERABLE status within 8 hours. INSERT 2

Condition A worst scenario is one train without AC power (i.e., no offsite power to the train and the associated EDG inoperable). In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining train by stabilizing the unit, and on restoring power to the affected train. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

Required Action A.1 is modified by a Note that requires the applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," to be entered for DC sources made inoperable by inoperable power distribution subsystems. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. Inoperability of a distribution system can result in loss of charging power to batteries and

BASES

ACTIONS

A.1 (continued)

eventual loss of DC power. This Note ensures that the appropriate attention is given to restoring charging power to batteries, if necessary, after loss of distribution systems.

B.1

With one or more AC vital buses inoperable, and a loss of function has not yet occurred, the remaining OPERABLE AC vital buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required AC vital bus must be restored to OPERABLE status within 8 hours by powering the bus from the associated inverter, via inverted 125 VDC voltage or Class 1E constant voltage transformer. INSERT 1

Condition B represents one or more AC vital buses without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation the unit is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining vital buses and restoring power to the affected vital bus.

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, that would have the Required Action Completion Times shorter than 8 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without adequate vital AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

BASES

ACTIONS

B.1 (continued)

The 8 hour Completion Time takes into account the importance to safety of restoring the AC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a DBA occurring during this period.

C.1

With one DC electrical power distribution subsystem inoperable, and a loss of function has not yet occurred, the remaining DC electrical power distribution subsystem is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the DC electrical power distribution subsystem must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger. INSERT 1

Condition C represents one DC electrical power distribution subsystem without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected train; and

Attachment 4

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Cross-Reference of TSTF-505 and DBNPS Technical Specifications

Table A4-1
Cross-Reference of TSTF-505 and DBNPS Technical Specifications

TSTF-505 TS Section / Condition Description	TSTF-505 TS / Required Action	DBNPS TS / Required Action	Disposition	Discussion and Attachment 1 Variation Reference
Completion Times	**1.3**	**1.3**		
Example 1.3-8	Example 1.3-8	Example 1.3-8	Incorporated example	No Variation
Reactor Protective System (RPS) Instrumentation	**3.3.1**	**3.3.1**		
B. Two channels inoperable.	3.3.1.B.1	3.3.1.B.1	RICT added	No Variation
Engineered Safety Feature Actuation System (ESFAS) Instrumentation	**3.3.5**	**3.3.5**		
A. One or more parameters with one channel inoperable	3.3.5.A.1	3.3.5.A.1	RICT added	The DBNPS TS title is “Safety Features Actuation System (SFAS) Instrumentation.” The associated LCO is different but maintains the same intent. Administrative Variation 2.3.1.1
Engineered Safety Feature Actuation System (ESFAS) Manual Initiation	**3.3.6**	**3.3.6**		
A. One or more ESFAS Functions with one channel inoperable	3.3.6.A.1	3.3.6.A.1	RICT added	The DBNPS TS title is “Safety Features Actuation System (SFAS) Manual Initiation.” The associated LCO is different but maintains the same intent. Administrative Variation 2.3.1.1

Table A4-1
Cross-Reference of TSTF-505 and DBNPS Technical Specifications

TSTF-505 TS Section / Condition Description	TSTF-505 TS / Required Action	DBNPS TS / Required Action	Disposition	Discussion and Attachment 1 Variation Reference
Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)	**3.3.8**	**3.3.8**		
A. One or more Functions with one channel per EDG inoperable	3.3.8.A.1	3.3.8.A.1	RICT added with LOF note	The associated LCO and Condition are different but maintain the same intent. Administrative Variations 2.3.1.1 and 2.3.1.6 (additional justification, including detail regarding the LOF, is provided in Enclosure 1)
B. One or more Functions with two or more channels per EDG inoperable	3.3.8.B.1	3.3.8.B.1	RICT added with LOF note	The associated LCO and Condition are different but maintain the same intent. Administrative Variations 2.3.1.1 and 2.3.1.6 (additional justification, including detail regarding the LOF, is provided in Enclosure 1)
Emergency Feedwater Initiation and Control (EFIC) System Instrumentation	**3.3.11**	**3.3.11**		
A. One or more Emergency Feedwater (EFW) Initiation, Main Steam Line Isolation, or Main Feedwater (MFW) Isolation Functions listed in Table 3.3.11-1 with one channel inoperable	3.3.11.A.2	3.3.11.A.1	No change – additional justification could not be provided demonstrating no loss of function	TSTF-505 changes not incorporated. Administrative Variation 2.3.1.5

Table A4-1
Cross-Reference of TSTF-505 and DBNPS Technical Specifications

TSTF-505 TS Section / Condition Description	TSTF-505 TS / Required Action	DBNPS TS / Required Action	Disposition	Discussion and Attachment 1 Variation Reference
B. One or more EFW Initiation, Main Steam Line Isolation, or MFW Isolation Functions listed in Table 3.3.11-1 with two channels inoperable	3.3.11.B.2	None	No change – DBNPS TS do not contain this RA	TSTF-505 changes not incorporated. Administrative Variation 2.3.1.2
	3.3.11.B.3	None	No change – DBNPS TS do not contain this RA	TSTF-505 changes not incorporated. Administrative Variation 2.3.1.2
C. One EFW Vector Valve Control channel inoperable	3.3.11.C.1	None	No change – DBNPS TS do not contain this RA	TSTF-505 changes not incorporated. Administrative Variation 2.3.1.2
Emergency Feedwater Initiation and Control (EFIC) Manual Initiation	**3.3.12**	**3.3.12**		
A. One or more EFIC Function(s) with one or both manual initiation switches inoperable in one actuation channel	3.3.12.A.1	None	No change – DBNPS TS do not contain this RA	TSTF-505 changes not incorporated. Administrative Variation 2.3.1.2
B. One or more EFIC Function(s) with one or both manual initiation switches inoperable in both actuation channels	3.3.12.B.1	3.3.12.A.1	RICT added	The DBNPS TS title is “Steam and Feedwater Rupture Control System (SFRCS) Manual Initiation.” The associated LCO, Condition, RA, and CT are different but maintain the same intent. Administrative Variation 2.3.1.1 (additional justification is provided in Enclosure 1)

Table A4-1
Cross-Reference of TSTF-505 and DBNPS Technical Specifications

TSTF-505 TS Section / Condition Description	TSTF-505 TS / Required Action	DBNPS TS / Required Action	Disposition	Discussion and Attachment 1 Variation Reference
Emergency Feedwater Initiation and Control (EFIC) Logic	**3.3.13**	**3.3.13**		
A. One or more channel A Functions inoperable with all channel B Functions OPERABLE OR One or more channel B Functions inoperable with all channel A Functions OPERABLE	3.3.13.A.1	3.3.13.A.1	RICT added	The DBNPS TS title is "Steam and Feedwater Rupture Control System (SFRCS) Actuation." The associated LCO, Condition, and RA are different but maintain the same intent. Administrative Variation 2.3.1.1 (additional justification is provided in Enclosure 1)
Emergency Feedwater Initiation and Control (EFIC) – Emergency Feedwater (EFW) – Vector Valve Logic	**3.3.14**	**None**		
A. One vector valve logic channel inoperable	3.3.14.A.1	None	No change – DBNPS TS do not contain this RA	TSTF-505 changes not incorporated. Administrative Variation 2.3.1.2
RCS Loops – MODE 3	**3.4.5**	**3.4.5**		
A. One RCS loop inoperable	3.4.5.A.1	3.4.5.A.1	No change – DBNPS RICT Program not applicable in Mode 3	TSTF-505 changes not incorporated. Administrative Variation 2.3.1.4

Table A4-1
Cross-Reference of TSTF-505 and DBNPS Technical Specifications

TSTF-505 TS Section / Condition Description	TSTF-505 TS / Required Action	DBNPS TS / Required Action	Disposition	Discussion and Attachment 1 Variation Reference
Pressurizer	**3.4.9**	**3.4.9**		
C. Capacity of pressurizer heaters [capable of being powered by emergency power supply] less than limit	3.4.9.C.1	3.4.9.C.1	RICT added	The associated Condition and RA are different but maintain the same intent. Administrative Variation 2.3.1.1 (additional justification is provided in Enclosure 1)
ECCS – Operating	**3.5.2**	**3.5.2**		
A. One low pressure injection (LPI) subsystem inoperable	3.5.2.A.1	3.5.2.A.1	RICT added	No Variation (additional justification is provided in Enclosure 1)
B. One or more trains inoperable for reasons other than Condition A	3.5.2.B.1	3.5.2.B.1	RICT added	No Variation (additional justification is provided in Enclosure 1)
Containment Air Locks	**3.6.2**	**3.6.2**		
C. One or more containment air locks inoperable for reasons other than Condition A or B	3.6.2.C.3	3.6.2.C.3	RICT added	No Variation (additional justification is provided in Enclosure 1)
Containment Isolation Valves	**3.6.3**	**3.6.3**		
A. One or more penetration flow paths with one containment isolation valve inoperable (applicable to penetration flow paths with two [or more] containment isolation valves)	3.6.3.A.1	3.6.3.A.1	RICT added	No Variation
	3.6.3.A.2	3.6.3.A.2	Added "following isolation"	No Variation
C. One or more penetration flow paths	3.6.3.C.1	3.6.3.C.1	RICT added	No Variation

Table A4-1
Cross-Reference of TSTF-505 and DBNPS Technical Specifications

TSTF-505 TS Section / Condition Description	TSTF-505 TS / Required Action	DBNPS TS / Required Action	Disposition	Discussion and Attachment 1 Variation Reference
with one containment isolation valve inoperable (applicable to penetration flow paths with only one containment isolation valve and a closed system)	3.6.3.C.2	3.6.3.C.2	Added "following isolation"	No Variation
Containment Spray and Cooling Systems	**3.6.6**	**3.6.6**		
A. One containment spray train inoperable	3.6.6.A.1	3.6.6.A.1	RICT added	No Variation (additional justification is provided in Enclosure 1)
C. One [required] containment cooling train inoperable	3.6.6.C.1	3.6.6.C.1	RICT added	No Variation (additional justification is provided in Enclosure 1)
D. One containment spray train and one [required] containment cooling train inoperable	3.6.6.D.1 / 3.6.6.D.2	3.6.6.D.1 / 3.6.6.D.2	RICT added	Administrative Variation 2.3.1.1 (additional justification is provided in Enclosure 1)
E. Two [required] containment cooling trains inoperable	3.6.6.E.1	3.6.6.E.1	RICT added	No Variation (additional justification is provided in Enclosure 1)
Main Steam Isolation Valves (MSIVs)	**3.7.2**	**3.7.2**		
A. One MSIV inoperable in MODE 1	3.7.2.A.1	3.7.2.A.1	RICT added	No Variation (additional justification is provided in Enclosure 1)
Atmospheric Vent Valves (AVVs)	**3.7.4**	**None**		
A. One required AVV [line] inoperable	3.7.4.A.1	None	No change – DBNPS TS do not contain this RA	TSTF-505 changes not incorporated. Administrative Variation 2.3.1.2

Table A4-1
Cross-Reference of TSTF-505 and DBNPS Technical Specifications

TSTF-505 TS Section / Condition Description	TSTF-505 TS / Required Action	DBNPS TS / Required Action	Disposition	Discussion and Attachment 1 Variation Reference
Emergency Feedwater (EFW) System	**3.7.5**	**3.7.5**		
A. One steam supply to turbine driven EFW pump inoperable OR One turbine driven EFW pump inoperable in MODE 3 following refueling	3.7.5.A.1	3.7.5.A.1	RICT added	The DBNPS TS title is "Emergency Feedwater (EFW)." The associated LCO, Condition, and RA are different but maintain the same intent. Administrative Variation 2.3.1.1
B. One EFW train inoperable [for reasons other than Condition A] in MODE 1, 2, or 3.	3.7.5.B.1	3.7.5.B.1	RICT added	The DBNPS TS title is "Emergency Feedwater (EFW)." The associated LCO is different but maintains the same intent. Administrative Variation 2.3.1.1
Component Cooling Water (CCW) System	**3.7.7**	**3.7.7**		
A. One CCW train inoperable	3.7.7.A.1	3.7.7.A.1	RICT added	The associated LCO, Condition, and RA are different but maintain the same intent. Administrative Variation 2.3.1.1
Service Water System (SWS)	**3.7.8**	**3.7.8**		
A. One SWS train inoperable	3.7.8.A.1	3.7.8.A.1	RICT added	The associated LCO, Condition, and RA are different but maintain the same intent. Administrative Variation 2.3.1.1

Table A4-1
Cross-Reference of TSTF-505 and DBNPS Technical Specifications

TSTF-505 TS Section / Condition Description	TSTF-505 TS / Required Action	DBNPS TS / Required Action	Disposition	Discussion and Attachment 1 Variation Reference
Ultimate Heat Sink (UHS)	**3.7.9**	**3.7.9**		
A. One or more cooling towers with one cooling tower fan inoperable	3.7.9.A.1	None	No change – DBNPS TS do not contain this RA	TSTF-505 changes not incorporated. Administrative Variation 2.3.1.2
AC Sources – Operating	**3.8.1**	**3.8.1**		
A. One [required] offsite circuit inoperable	3.8.1.A.3	3.8.1.A.3	RICT added	No Variation
B. One [required] DG inoperable	3.8.1.B.4	3.8.1.B.4	RICT added	The associated Condition, RA, and CT are different but maintain the same intent. Administrative Variation 2.3.1.1
C. Two [required] offsite circuits inoperable	3.8.1.C.2	3.8.1.C.2	RICT added	No Variation
D. One [required] offsite circuit inoperable. <u>AND</u> One [required] DG inoperable	3.8.1.D.1 / 3.8.1.D.2	3.8.1.D.1 / 3.8.1.D.2	RICT added	The associated Condition and RA are different but maintain the same intent. Administrative Variation 2.3.1.1
F. One [required] [automatic load sequencer] inoperable	3.8.1.F.1	3.8.1.G.1	RICT added	The associated Condition and RA are different but maintain the same intent. Administrative Variation 2.3.1.1

Table A4-1
Cross-Reference of TSTF-505 and DBNPS Technical Specifications

TSTF-505 TS Section / Condition Description	TSTF-505 TS / Required Action	DBNPS TS / Required Action	Disposition	Discussion and Attachment 1 Variation Reference
DC Sources – Operating	**3.8.4**	**3.8.4**		
A. One [or two] battery charger[s on one train] inoperable	3.8.4.A.3	3.8.4.A.3	RICT added	The associated LCO, Condition, and CT are different but maintain the same intent. Administrative Variation 2.3.1.1
B. One [or two] batter[y][ies on one train] inoperable	3.8.4.B.1	None	No change – DBNPS TS do not contain this RA	TSTF-505 changes not incorporated. Administrative Variation 2.3.1.2
C. One DC electrical power subsystem inoperable for reasons other than Condition A [or B]	3.8.4.C.1	3.8.4.B.1	RICT added	The associated LCO, Condition, and RA are different but maintain the same intent. Administrative Variation 2.3.1.1
Inverters – Operating	**3.8.7**	**3.8.7**		
A. One [required] inverter inoperable	3.8.7.A.1	3.8.7.A.1	RICT added	The associated LCO is different but maintains the same intent. Administrative Variation 2.3.1.1
Distribution Systems – Operating	**3.8.9**	**3.8.9**		
A. One or more AC electrical power distribution subsystems inoperable	3.8.9.A.1	3.8.9.A.1	RICT added	The associated LCO is different but maintains the same intent. Administrative Variation 2.3.1.1

Table A4-1
Cross-Reference of TSTF-505 and DBNPS Technical Specifications

TSTF-505 TS Section / Condition Description	TSTF-505 TS / Required Action	DBNPS TS / Required Action	Disposition	Discussion and Attachment 1 Variation Reference
B. One or more AC vital buses inoperable	3.8.9.B.1	3.8.9.B.1	RICT added	The associated LCO, RA, and CT are different but maintain the same intent. Administrative Variation 2.3.1.1
C. One or more DC electrical power distribution subsystems inoperable	3.8.9.C.1	3.8.9.C.1	RICT added	The associated LCO and Condition are different but maintain the same intent. Administrative Variation 2.3.1.1
Programs and Manuals	**5.5**	**5.5**		
Risk Informed Completion Time Program	5.5.18	5.5.20	Program added	Minor change added to Program wording. Administrative Variations 2.3.1.7

Attachment 5

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Evaluation of Instrumentation and Control Systems

Information Supporting Redundancy and Diversity

This attachment provides justification for TS 3.3, “Instrumentation,” that defense-in-depth objectives are met. Defense-in-depth objectives state that there is at least one redundant or diverse means (through other automatic features or manual action) to accomplish necessary safety functions during application of the RICT. This attachment identified the components available to respond to identified accident conditions located within UFSAR Section 15.

Based on the information laid out in Section 3.1.2.3 of TSTF-505-A, the following Instrumentation TS sections are included in this LAR for DBNPS:

1. Reactor Protection System (RPS) Instrumentation - TS Section 3.3.1
2. Safety Features Actuation System (SFAS) Instrumentation – TS Section 3.3.5
3. Emergency Diesel Generator (EDG) Loss of Power Start (LOPS) Instrumentation – TS Section 3.3.8
4. Steam and Feedwater Rupture Control System (SFRCS) Instrumentation – TS Section 3.3.11

DBNPS TS 3.3 Instrumentation Limiting Conditions for Operation (LCOs) were developed to ensure that DBNPS maintains necessary redundancy and diversity of systems, structures, and components (SSCs), SFAS (TS 3.3.5) and SFRCS (TS 3.3.11) comply with the single failure design criterion as defined in IEEE 279-1971, RPS (TS 3.3.1) complies with the single failure design criterion as defined in IEEE 279-1968, and the diversity requirements as defined in Appendix A, “General Design Criteria for Nuclear Power Plants” to Part 50 of 10 CFR, GDC-22, “Protection System Independence.”

Included below is a description of the redundant and diverse means available to mitigate accidents that each identified instrumentation and control function defined in TS Section 3.3 is designed to prevent.

Tables are provided for each TS Instrumentation section to identify the Updated Final Safety Analysis Report (UFSAR) transient/accident that credits the instrumentation and control function. For each instrumentation and control function, diverse instrumentation is identified to demonstrate that at least one diverse means is available to accomplish the associated safety function. The UFSAR and engineering judgement were used to identify transient/accidents that would challenge a response from the listed instrumentation. The list of potential accidents and redundant instrumentation are best estimates and are in no way exhaustive.

A.1. Reactor Protection Instrumentation - TS Section 3.3.1

The RPS is comprised of four separate redundant protection channels. The generation of any two trip signals in any of the four RPS channels will subsequently trip the reactor. Each of the 4 channels of RPS contain a Reactor Trip Module (RTM) which receives trip signals from its own measurement channels and transmits these signals to the other three RPS channels. The reactor is finally tripped by opening circuit breakers that interrupt the power supply to the Control Rod Drives (CRDs).

The RPS design creates defense-in-depth from the redundancy of the channels for each trip system.

The diverse inputs causing a trip of the RPS are: (UFSAR Chapter 7.2, UFSAR Chapter 15, TS Table 3.3.1-1 and Bases B 3.3.1):

- 1.a. Neutron Flux – High Setpoint
  - 2 out of 4 RPS Channels to trip
- 1.b. Neutron Flux – Low Setpoint
  - 2 out of 4 RPS Channels to trip
- 2. Reactor Coolant (RC) High Temperature
  - 2 out of 4 RPS Channels to trip
- 3. Reactor Coolant (RC) High Pressure
  - 2 out of 4 RPS Channels to trip
- 4. Reactor Coolant (RC) Low Pressure
  - 2 out of 4 RPS Channels to trip
- 5. Reactor Coolant (RC) Pressure – Temperature
  - 2 out of 4 RPS Channels to trip
- 6. Containment High Pressure
  - 2 out of 4 RPS Channels to trip
- 7. High Flux/Number of Reactor Coolant Pumps On
  - 2 out of 4 RPS Channels to trip
- 8. Flux – ΔFlux – Flow
  - 2 out of 4 RPS Channels to trip
- 9. Shutdown Bypass High Pressure
  - 2 out of 4 RPS Channels to trip

Table A.1: RPS Instrumentation Redundancy and Diversity			
Function	**UFSAR Section**	**Transient Accident**	**Redundant/Diverse Instrumentation[1]**
1a. Neutron Flux – High Setpoint (2 out of 4 RPS Channels to trip)	15.2.1	Uncontrolled Control Rod Assembly Group Withdrawal from a Subcritical Condition (Startup Accident)	1) Automatic Initiation: - *RPS Neutron Flux – High Setpoint* - RPS Reactor Coolant (RC) High Pressure 2) Manual Trip
	15.2.2	Uncontrolled Control Rod Assembly Group Withdrawal at Power	1) Automatic Initiation: - *RPS Neutron Flux – High Setpoint* - RPS Reactor Coolant (RC) High Pressure 2) Manual Trip
	15.2.10	Excessive Heat Removal Due to Feedwater System Malfunction	1) Automatic Initiation: - *RPS Neutron Flux – High Setpoint* - RPS Reactor Coolant (RC) Low Pressure 2) Manual Trip
	15.2.11	Excessive Load Increase	1) Automatic Initiation: - *RPS Neutron Flux – High Setpoint* - RPS Reactor Coolant (RC) Low Pressure 2) Manual Trip
	15.4.3	CRA Ejection Accident	1) Automatic Initiation: - *RPS Neutron Flux – High Setpoint* - RPS Reactor Coolant (RC) High Pressure 2) Manual Trip
1b. Neutron Flux – Low Setpoint (2 out of 4 RPS Channels to trip)	N/A	Transition to/from mode 2 and 3 evolutions	1) Automatic Initiation: - *RPS Neutron Flux – Low Setpoint* - RPS Shutdown Bypass High Pressure 2) Manual Trip

Table A.1: RPS Instrumentation Redundancy and Diversity			
Function	**UFSAR Section**	**Transient Accident**	**Redundant/Diverse Instrumentation[1]**
2. Reactor Coolant (RC) High Temperature (2 out of 4 RPS Channels to trip)	15.2.4	Makeup and Purification System Malfunction	1) Automatic Initiation - *RPS Reactor Coolant (RC) High Temperature* - RPS Reactor Coolant (RC) High Pressure - RPS Reactor Coolant (RC) Pressure – Temperature 2) Manual Trip
	15.2.8	Loss of Normal Feedwater	1) Automatic Initiation - *RPS Reactor Coolant (RC) High Temperature* - RPS Reactor Coolant (RC) High Pressure - RPS Reactor Coolant (RC) Pressure – Temperature 2) Manual Trip
3. Reactor Coolant (RC) High Pressure (2 out of 4 RPS Channels to trip)	15.2.1	Uncontrolled Control Rod Assembly Group Withdrawal from a Subcritical Condition (Startup Accident)	1) Automatic Initiation: - RPS Neutron Flux – High Setpoint - *RPS Reactor Coolant (RC) High Pressure* 2) Manual Trip
	15.2.2	Uncontrolled Control Rod Assembly Group Withdrawal at Power	1) Automatic Initiation: - RPS Neutron Flux – High Setpoint - *RPS Reactor Coolant (RC) High Pressure* 2) Manual Trip
	15.2.4	Makeup and Purification System Malfunction	1) Automatic Initiation - RPS Reactor Coolant (RC) High Temperature - *RPS Reactor Coolant (RC) High Pressure* - RPS Reactor Coolant (RC) Pressure – Temperature 2) Manual Trip

Table A.1: RPS Instrumentation Redundancy and Diversity			
Function	**UFSAR Section**	**Transient Accident**	**Redundant/Diverse Instrumentation[1]**
	15.2.8	Loss of Normal Feedwater	1) Automatic Initiation - RPS Reactor Coolant (RC) High Temperature - *RPS Reactor Coolant (RC) High Pressure* - RPS Reactor Coolant (RC) Pressure – Temperature 2) Manual Trip
	15.4.3	CRA Ejection Accident	1) Automatic Initiation: - RPS Neutron Flux – High Setpoint - *RPS Reactor Coolant (RC) High Pressure* 2) Manual Trip
4. Reactor Coolant (RC) Low Pressure (2 out of 4 RPS Channels to trip)	15.2.10	Excessive Heat Removal Due to Feedwater System Malfunction	1) Automatic Initiation: - RPS Neutron Flux – High Setpoint - *RPS Reactor Coolant (RC) Low Pressure* 2) Manual Trip
	15.2.11	Excessive Load Increase	1) Automatic Initiation: - RPS Neutron Flux – High Setpoint - *RPS Reactor Coolant (RC) Low Pressure* 2) Manual Trip
	15.4.2	Steam Generator Tube Rupture	1) Automatic Initiation: - *RPS Reactor Coolant (RC) Low Pressure* 2) Manual Trip
	15.4.4	Steam Line Break	1) Automatic Initiation: - *RPS Reactor Coolant (RC) Low Pressure* - RPS Neutron Flux – High Setpoint 2) Manual Trip

Table A.1: RPS Instrumentation Redundancy and Diversity

Function	UFSAR Section	Transient Accident	Redundant/Diverse Instrumentation[1]
	15.4.5	Break in Instrument Lines or Lines from Primary System that Penetrate Containment	1) Automatic Initiation: - *RPS Reactor Coolant (RC) Low Pressure* 2) Manual Trip
	15.4.6	Major Rupture of Pipes Containing Reactor Coolant up to and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Loss-of-Coolant Accident)	1) Automatic Initiation: - *RPS Reactor Coolant (RC) Low Pressure* - RPS Containment High Pressure
5. Reactor Coolant (RC) Pressure – Temperature (2 out of 4 RPS Channels to trip)	15.2.4	Makeup and Purification System Malfunction	1) Automatic Initiation - RPS Reactor Coolant (RC) High Temperature - RPS Reactor Coolant (RC) High Pressure - *RPS Reactor Coolant (RC) Pressure – Temperature* 2) Manual Trip
	15.2.8	Loss of Normal Feedwater	1) Automatic Initiation - RPS Reactor Coolant (RC) High Temperature - RPS Reactor Coolant (RC) High Pressure - *RPS Reactor Coolant (RC) Pressure – Temperature* 2) Manual Trip

Table A.1: RPS Instrumentation Redundancy and Diversity			
Function	**UFSAR Section**	**Transient Accident**	**Redundant/Diverse Instrumentation[1]**
6. Containment High Pressure (2 out of 4 RPS Channels to trip)	15.4.6	Major Rupture of Pipes Containing Reactor Coolant up to and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Loss-of-Coolant Accident)	1) Automatic Initiation: - RPS Reactor Coolant (RC) Low Pressure - *RPS Containment High Pressure*
7. High Flux/Number of Reactor Coolant Pumps On (2 out of 4 RPS Channels to trip)	15.2.5	Loss of Forced Reactor Coolant Flow (Partial, Complete, and Single Reactor Coolant Pump Locked Rotor)	1) Automatic Initiation: - *RPS High Flux/Number of Reactor Coolant Pumps On* - RPS Flux – ΔFlux – Flow 2) Manual Trip
8. Flux – ΔFlux – Flow (2 out of 4 RPS Channels to trip)	15.2.5	Loss of Forced Reactor Coolant Flow (Partial, Complete, and Single Reactor Coolant Pump Locked Rotor)	1) Automatic Initiation: - RPS High Flux/Number of Reactor Coolant Pumps On - *RPS Flux – ΔFlux – Flow* 2) Manual Trip
9. Shutdown Bypass High Pressure (2 out of 4 RPS Channels to trip)	N/A	Transition to/from mode 2 and 3 evolutions	1) Automatic Initiation: - *RPS Shutdown Bypass High Pressure* - RPS Neutron Flux – Low Setpoint 2) Manual Trip

Note 1: Instrumentation in italics is the affected function crediting redundant channels.

A.2. Safety Features Actuation System (SFAS) Instrumentation – TS Section 3.3.5

The SFAS instrumentation initiates necessary safety systems in the event of a loss-of-coolant accident (LOCA) and a main steam line break (MSLB). SFAS actuates various systems at different levels based on which parameter set points are exceeded (see Notes 2 through 6 of Table A.2).

SFAS is comprised of four channels monitoring each actuation parameter which send signals to logic channels with a two-out-of-four logic. If any 2 out of the 4 logic channels exceed setpoint for each of the SFAS parameters, the relevant level of SFAS will actuate.

The SFAS design creates defense-in-depth from the redundancy of initiation systems for the various initiation functions.

Diverse SFAS Initiation Parameters:

- 1. Reactor Coolant System Pressure – Low
    - 2 out of 4 SFAS Channels to initiate
- 2. Reactor Coolant System Pressure – Low Low
    - 2 out of 4 SFAS Channels to initiate
- 3. Containment Pressure – High
    - 2 out of 4 SFAS Channels to initiate
- 4. Containment Pressure – High High
    - 2 out of 4 SFAS Channels to initiate
- 5. Borated Water Storage Tank Level – Low Low
    - 2 out of 4 SFAS Channels to allow manual transfer

Table A.2: SFAS Instrumentation Redundancy and Diversity				
Function	**SFAS Actuation Level**	**UFSAR Section**	**Transient Accident**	**Redundant/Diverse Instrumentation[1]**
1. Reactor Coolant System Pressure – Low (2 out of 4 SFAS Channels to Initiate)	Level 1[2] Level 2[3]	15.3.1	Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates Emergency Core Cooling (Small LOCA)	1) Automatic Initiation: - *SFAS Reactor Coolant System Pressure – Low* - SFAS Containment Pressure – High 2) Manual Initiation
		15.4.6	Major Rupture of Pipes Containing Reactor Coolant Up To and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Large LOCA)	1) Automatic Initiation: - *SFAS Reactor Coolant System Pressure – Low* - SFAS Containment Pressure – High 2) Manual Initiation
		15.4.4	Steam Line Break (Inside Containment)	1) Automatic Initiation: - *SFAS Reactor Coolant System Pressure – Low* - SFAS Containment Pressure – High 2) Manual Initiation
		15.4.2	Steam Generator Tube Rupture (SGTR)	1) Automatic Initiation: - *SFAS Reactor Coolant System Pressure – Low* 2) Manual Initiation
2. Reactor Coolant System Pressure – Low Low (2 out of 4 SFAS Channels to Initiate)	Level 3[4]	15.4.6	Major Rupture of Pipes Containing Reactor Coolant Up To and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Large LOCA)	1) Automatic Initiation: - *SFAS Reactor Coolant System Pressure – Low Low* - SFAS Containment Pressure – High 2) Manual Initiation
3. Containment Pressure – High (2 out of 4 SFAS Channels to Initiate)	Level 1[2] Level 2[3] Level 3[4]	15.3.1	Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates Emergency Core Cooling (Small LOCA)	1) Automatic Initiation: - SFAS Reactor Coolant System Pressure – Low - *SFAS Containment Pressure – High* 2) Manual Initiation

Table A.2: SFAS Instrumentation Redundancy and Diversity				
Function	**SFAS Actuation Level**	**UFSAR Section**	**Transient Accident**	**Redundant/Diverse Instrumentation[1]**
		15.4.6	Major Rupture of Pipes Containing Reactor Coolant Up To and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Large LOCA)	1) Automatic Initiation: - SFAS Reactor Coolant System Pressure – Low - SFAS Reactor Coolant System Pressure – Low Low - *SFAS Containment Pressure – High* 2) Manual Initiation
		15.4.4	Steam Line Break (Inside Containment)	1) Automatic Initiation: - Reactor Coolant System Pressure – Low - *Containment Pressure – High* 2) Manual Initiation
4. Containment Pressure – High High (2 out of 4 SFAS Channels to Initiate)	Level 4[5]	15.4.6	Major Rupture of Pipes Containing Reactor Coolant Up To and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Large LOCA)	1) Manual Initiation
5. Borated Water Storage Tank Level – Low Low (2 out of 4 SFAS Channels to Initiate)	Level 5[6]	15.3.1	Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates Emergency Core Cooling (Small LOCA)	1) Manual Initiation
		15.4.6	Major Rupture of Pipes Containing Reactor Coolant Up To and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant System (Large LOCA)	1) Manual Initiation

Note 1: Instrumentation in italics is the affected function crediting redundant channels.

Note 2: SFAS Actuation Level 1 includes isolation of Containment Purge and Exhaust, Containment Sampling, and Control Room Normal Ventilation Systems as well as actuation of Station Emergency Ventilation System.

Note 3: SFAS Actuation Level 2 includes initiation of HPI, CCW, SWS, CAC coolers, EDGs, and closure of level 2 containment isolations.

Note 4: SFAS Actuation Level 3 includes initiation of LPI and closure of Level 3 containment isolations.

Note 5: SFAS Actuation Level 4 includes initiation of containment spray and closure of Level 4 containment isolations.

Note 6: SFAS Actuation Level 5 enables the use of recirculation permissive.

A.3. Emergency Diesel Generator (EDG) Loss of Power Start (LOPS) Instrumentation – TS Section 3.3.8

The purpose of the EDG LOPS instrumentation is to ensure that the EDGs are ready to be actuated by the SFAS system in the event of an accident with loss of offsite power or degraded voltage condition in the switchyard. The EDG LOPS instrumentation monitors for degraded and loss of voltage conditions on the EDGs via undervoltage and auxiliary relays, and consists of a total of 4 channels per function. Two undervoltage relays and an auxiliary relay per essential bus are associated with a channel. There are 8 undervoltage relays and 4 auxiliary relays per function, for a total of 16 undervoltage relays and 8 auxiliary relays. The actuation of both auxiliary relays on an electrical bus (two-out-of-two logic) will disconnect the offsite source and generate an EDG LOPS signal.

The EDG LOPS design creates defense-in-depth from the redundancy of trip systems for the Trip Function.

Diverse EDG LOPS trip systems:

- 1. Loss of Voltage (LOV) Function
    - 8 LOV relays and 4 auxiliary relays. 1 auxiliary relay per channel and 2 channels per electrical bus.
- 2. Degraded Voltage (DV) Function
    - 8 DV relays and 4 auxiliary relays. 1 auxiliary relay per channel and 2 channels per electrical bus.

Table A.3: EDG LOPS Instrumentation Redundancy and Diversity					
Function	**Safety Function**	**UFSAR Section**	**Transient Accident**	**Redundant Available Channel Logic**	**Diverse Instrumentation/Initiation**
a. Loss of Voltage Function (2 electrical buses with 2 channels each with 2 LOV relays per channel. 1 LOV relay signal on both channels per bus create an EDG LOPS signal)	Disconnect offsite source, load shed the affected essential bus, and generate EDG LOPS signal	15.2.7	Loss of External Electrical Load and/or Turbine Trip (Loss of Offsite Power)	1 available LOV relay on both channels (No redundancy for auxiliary relays).	1) Automatic Initiation *- EDG LOPS Loss of Voltage Function* - EDG LOPS Degraded Voltage Function
b. Degraded Voltage Function (2 electrical buses with 2 channels each with 2 DV relays per channel. 1 DV relay signal on both channels per bus create an EDG LOPS signal)	Disconnect offsite source, load shed the affected essential bus, and generate EDG LOPS signal	15.2.7	Loss of External Electrical Load and/or Turbine Trip (Loss of Offsite Power)	1 available DV relay on both channels (No redundancy for auxiliary relays).	1) Automatic Initiation - EDG LOPS Loss of Voltage Function *- EDG LOPS Degraded Voltage Function*

Note 1: Instrumentation in italics is the affected function crediting redundant channels.

A.4. Steam and Feedwater Rupture Control System (SFRCS) Instrumentation – TS Section 3.3.11

The purpose of the SFRCS instrumentation is to initiate Auxiliary Feedwater (AFW), and in some circumstances, isolate the steam generators in the event of an MSLB, main feedwater line break, low level in the steam generators, or a loss of all four reactor coolant pumps. For each SFRCS function excluding loss of the RCPs, each parameter has 4 logic channels in which 2 out of the 4 logic channels must be actuated to initiate AFW and/or isolate the steam generators. For loss of RCPs, 2 out of 4 logic channels on the same actuation channel that receive undercurrent signals from all 4 RCPs must actuate to initiate SFRCS and start AFW.

The SFRCS Instrumentation design creates defense-in-depth from the redundancy of initiation systems for the various initiation functions.

Diverse SFRCS instrumentation inputs:

- 1. Main Steam Line Pressure - Low
  - 8 instrument Channels, 4 per Main Steam Line (MSL). 4 logic Channels and 2 actuation Channels.
- 2. Feedwater/Steam Generator Differential Pressure – High
  - 8 instrument Channels, 4 per Main Feedwater Line (MFWL). 4 logic Channels and 2 actuation Channels.
- 3. Steam Generator Level – Low
  - 8 instrument Channels, 4 per Steam Generator (SG). 4 logic Channels and 2 actuation Channels.
- 4. Loss of RCPs
  - 4 instrument Channels, which monitor each Reactor Coolant Pump (RCP). 4 logic Channels and 2 actuation Channels.

Table A.4 – SFRCS Instrumentation Redundancy and Diversity

Function	Safety Function	Accident Class	Redundant Available Channel Logic	Diverse Instrumentation/Initiation
1. Main Steam Line Pressure - Low (2 out of 2 logic channels on 1 out of 2 actuation channels on the SG to initiate)	Start Auxiliary Feedwater (AFW)[1]	Main Steam Line Break (MSLB)	2 out of 3 of available Main Steam Line Pressure – Low logic channels on the SG.	1) Automatic Initiation - SFRCS Feedwater/Steam Generator Differential Pressure – High 2) Manual Initiation
		Main Feedwater Line Break (MFWLB)[3]		
	Isolate SG[2]	Main Steam Line Break (MSLB)		
		Main Feedwater Line Break (MFWLB)[3]		
2. Feedwater/Steam Generator Differential Pressure – High	Start AFW (2 out of 4 logic channels on 1 out of 2 actuation channels on the FWL to initiate)	MSLB	2 out of 3 available Feedwater/Steam Generator Differential Pressure – High logic channels on the FWL.	1) Automatic Initiation - SFRCS Main Steam Line Pressure - Low 2) Manual Initiation

Table A.4 – SFRCS Instrumentation Redundancy and Diversity

Function	Safety Function	Accident Class	Redundant Available Channel Logic	Diverse Instrumentation/Initiation
2. Feedwater/Steam Generator Differential Pressure – High	Isolate SGs (2 out of 4 logic channels on 1 out of 2 actuation channels on the FWL to initiate)	MFWLB[4]	2 out of 3 available Feedwater/Steam Generator Differential Pressure – High logic channels on the FWL.	1) Automatic Initiation - SFRCS Steam Generator Level – Low 2) Manual Initiation
		Loss of Feedwater (LOFW)		1) Automatic Initiation - SFRCS Steam Generator Level – Low 2) Manual Initiation
		Loss of Offsite Power (LOOP)		1) Automatic Initiation - SFRCS Steam Generator Level – Low 2) Manual Initiation
		MSLB	2 out of 3 available Feedwater/Steam Generator Differential Pressure – High logic channels on the FWL.	1) Automatic Initiation - SFRCS Main Steam Line Pressure - Low 2) Manual Initiation
		MFWLB[4]		1) Automatic Initiation - SFRCS Steam Generator Level – Low 2) Manual Initiation
		LOFW		1) Automatic Initiation - SFRCS Steam Generator Level – Low 2) Manual Initiation
		LOOP		1) Automatic Initiation - SFRCS Steam Generator Level – Low 2) Manual Initiation
3. Steam Generator Level – Low	Start AFW	MFWLB		1) Automatic Initiation - SFRCS Main Steam Line Pressure – Low[3]

Table A.4 – SFRCS Instrumentation Redundancy and Diversity

Function	Safety Function	Accident Class	Redundant Available Channel Logic	Diverse Instrumentation/Initiation
	(2 out of 4 logic channels on 1 out of 2 actuation channels on the SG to initiate)		2 out of 3 available Steam Generator Level – Low logic channels on the SG.	- SFRCS Feedwater/Steam Generator Differential Pressure – High[4] 2) Manual Initiation
		LOFW		1) Automatic Initiation - SFRCS Feedwater/Steam Generator Differential Pressure – High 2) Manual Initiation
		LOOP		1) Automatic Initiation - SFRCS Feedwater/Steam Generator Differential Pressure – High - SFRCS Loss of RCPs 2) Manual Initiation
4. Loss of RCPs (2 out of 4 logic channels on the same actuation channel any of the 4 RCPs to initiate)	Start AFW	Loss of RCS Flow	2 out of 3 available Loss of RCPs logic channels.	1) Manual Initiation
		LOOP		1) Automatic Initiation - SFRCS Feedwater/Steam Generator Differential Pressure – High - Steam Generator Level – Low 2) Manual Initiation

Note 1: Align AFW to non-affected steam generator.
Note 2: Isolates feedwater and main steam on affected steam generator.
Note 3: Downstream of check valve isolating steam generator.
Note 4: Upstream of check valve isolating steam generator.

Enclosure 1

Davis-Besse Nuclear Power Station

License Amendment Request

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b"

List of Revised Required Actions to Corresponding PRA Functions

List of Revised Required Actions to Corresponding PRA Functions

1.0 Introduction

Section 4.0, "Limitations and Conditions", Item 2 of the NRC Final Safety Evaluation 06-09 (Reference 1) for Nuclear Energy Institute (NEI) Topical Report NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines", Revision 0 (Reference 2), identifies the following needed content:

- The license amendment request (LAR) will provide identification of the TS Limiting Conditions for Operation (LCOs) and action requirements to which the RMTS will apply.
- The LAR will provide a comparison of the TS functions to the PRA modeled functions of the structures, systems, and components (SSCs) subject to those LCO actions.
- The comparison should justify that the scope of the PRA model, including applicable success criteria such as number of SSCs required, flow rate, etc., are consistent with licensing basis assumptions (i.e., 50.46 [Emergency Core Cooling System (ECCS)] flowrates) for each of the TS requirements, or an appropriate disposition or programmatic restriction will be provided.

This enclosure provides confirmation that the Davis-Besse Nuclear Power Station (DBNPS) PRA models include the necessary scope of SSCs and their functions to address each proposed application of the Risk-Informed Completion Time (RICT) Program to the proposed scope TS LCO Conditions, and provides the information requested for Section 4.0, Item 2 of the NRC Final Safety Evaluation. The scope of the comparison includes each of the TS LCO conditions and associated required actions within the scope of the RICT Program.

Table E1-1 below lists each TS LCO Condition to which the RICT Program is proposed to be applied and documents the following information regarding the TSs with the associated safety analyses, the analogous PRA functions, and the results of the comparison:

- Column "Tech Spec Description": Lists all of the LCOs and condition statements within the scope of the RICT Program.
- Column "SSCs Covered by TS LCO Condition and Applicable Mode(s)": List the SSCs addressed by each action requirement. Note that SSCs not applicable to the DBNPS RICT Program are not listed.
- Column "Modeled in PRA": Indicates whether the SSCs addressed by the TS LCO Condition are included in the PRA.
- Column "Function Covered by TS LCO Condition": Lists a summary of the required functions from the design basis analyses.
- Column "Design Success Criteria": Provides a summary of the success criteria from the design basis analyses.

- Column "PRA Success Criteria": List the function success criteria modeled in the PRA.

- Column "Comments": Provides the justification or resolution to address any inconsistencies between the TS and PRA functions regarding the scope of SSCs and the success criteria. Where the PRA scope of SSCs is not consistent with the TS, additional information is provided to describe how the LCO condition can be evaluated using appropriate surrogate events. Differences in the success criteria for TS functions are addressed to demonstrate the PRA criteria provide a realistic estimate of the risk of the TS condition as required by NEI 06-09-A, Revision 0 (Reference 2).

The corresponding SSCs for each TS LCO and the associated TS functions are identified and compared to the PRA. This description also includes the design success criteria and the applicable PRA success criteria. Any differences between the scope or success criteria are described in the table. Scope differences are justified by identifying appropriate surrogate events which permit a risk evaluation to be completed using the Configuration Risk Management Program tool for the RICT Program. Differences in success criteria typically arise due to the requirement in the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) RA-Sa-2009 PRA Standard (hereafter "ASME/ANS PRA Standard") (Reference 3) to make PRAs realistic rather than bounding, whereas design basis criteria are necessarily conservative and bounding. The use of realistic success criteria is necessary to conform to capability Category II of the ASME/ANS PRA standard as required by NEI 06-09-A, Revision 0.

Examples of calculated RICT are provided in Table E1-2 for each individual condition to which the RICT applies (assuming no other SSCs modeled in the PRA are unavailable). These example calculations demonstrate the scope of the SSCs covered by TSs modeled in the PRA. Note that the more limiting of the core damage frequency (CDF) and large early release frequency (LERF) RICT result is shown.

Following implementation of the RICT Program, the actual RICT values will be calculated using the actual plant configuration and the current revision of the PRA model representing the as-built, as-operated condition of the plant, as required by NEI 06-09-A (reference 2) and the NRC Final Safety Evaluation. The actual RICT values may differ from the RICTs presented in this enclosure.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.3.1.B.1	Reactor Protection System (RPS) Instrumentation Two channels inoperable	1a. Neutron Flux – High Setpoint (4 RPS Channels)	No	Initiate reactor trip when monitored parameter reaches setpoint	With one Neutron Flux – High Setpoint channel inoperable (INOP), any 2-out-of-3 operable Neutron Flux – High Setpoint channels.	None	A single point estimate cause failure event addresses the failure of the RPS scram function. It is suggested that the RPS event model should be expanded in the configuration risk monitor program (CRMP) model prior to implementation (see Note 1)
		1b. Neutron Flux – Low Setpoint (4 RPS Channels)	No	Initiate reactor trip when monitored parameter reaches setpoint	With one Neutron Flux – Low Setpoint channel INOP, any 2-out-of-3 operable Neutron Flux – Low Setpoint channels.	None	
		2. RC High Temperature (4 RPS Channels)	No	Initiate reactor trip when monitored parameter reaches setpoint	With one RC High Temperature channel INOP, any 2-out-of-3 operable RC High Temperature channels.	None	
		3. RC High Pressure (4 RPS Channels)	No	Initiate reactor trip when monitored parameter reaches setpoint	With one RC High Pressure channel INOP, any 2-out-of-3 operable RC High Pressure channels.	None	
		4. RC Low Pressure (4 RPS Channels)	No	Initiate reactor trip when monitored parameter reaches setpoint	With one RC Low Pressure channel INOP, any 2-out-of-3 operable RC Low Pressure channels.	None	
		5. RC Pressure – Temperature (4 RPS Channels)	No	Initiate reactor trip when monitored parameter reaches setpoint	With one RC Pressure – Temperature channel INOP, any 2-out-of-3 operable RC Pressure – Temperature channels.	None	

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
		6. Containment High Pressure (4 RPS Channels)	No	Initiate reactor trip when monitored parameter reaches setpoint	With one Containment High Pressure channel INOP, any 2-out-of-3 operable Containment High Pressure channels.	None	
		7. High Flux/Number of Reactor Coolant Pumps On (4 RPS Channels)	No	Initiate reactor trip when monitored parameter reaches setpoint	With one High Flux/Number of Reactor Coolant Pumps On channel INOP, any 2-out-of-3 operable High Flux/Number of Reactor Coolant Pumps On channels.	None	A single point estimate cause failure event addresses the failure of the RPS scram function. The RPS event model will be expanded in the CRMP model prior to implementation (see Note 1)
		8. Flux – ΔFlux – Flow (4 RPS Channels)	No	Initiate reactor trip when monitored parameter reaches setpoint	With one Flux – ΔFlux – Flow channel INOP, any 2-out-of-3 operable Flux – ΔFlux – Flow channels.	None	
		9. Shutdown Bypass High Pressure (4 RPS Channels)	No	Initiate reactor trip when monitored parameter reaches setpoint	With one Shutdown Bypass High Pressure channel INOP, any 2-out-of-3 operable Shutdown Bypass High Pressure channels.	None	
3.3.5.A.1	Safety Features Actuation System (SFAS) Instrumentation One or more Parameters with one channel	1. Reactor Coolant System Pressure – Low (4 SFAS channels)	Yes	Initiate SFAS Levels 1 and 2 (see Note 2)	With one Reactor Coolant System Pressure – Low channel INOP, any 2-out-of-3 operable Reactor Coolant System Pressure – Low channels.	Same as design success criteria	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
	inoperable.	2. Reactor Coolant System Pressure – Low Low (4 SFAS channels)	Yes	Initiate SFAS Level 3 (see Note 2)	With one Reactor Coolant System Pressure – Low Low channel INOP, 2-out-of-3 operable Reactor Coolant System Pressure – Low Low channels.	Same as design success criteria	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
		3. Containment Pressure – High (4 SFAS channels)	Yes	Initiate SFAS Levels 1, 2, and 3 (see Note 2)	With one Containment Pressure – High channel INOP, any 2-out-of-3 operable Containment Pressure – High channels.	Same as design success criteria	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
		4. Containment Pressure – High High (4 SFAS channels)	Yes	Initiate SFAS Level 4 (see Note 2)	With one Containment Pressure – High High channel INOP, any 2-out-of-3 operable Containment Pressure – High High channels.	Same as design success criteria	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
		5. Borated Water Storage Tank Level – Low Low (4 SFAS channels)	Yes	Initiate SFAS Level 5 (see Note 2)	With one Borated Water Storage Tank Level – Low Low channel INOP, any 2-out-of-3 operable Borated Water Storage Tank Level – Low Low channels.	Same as design success criteria	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.3.6.A.1	SFAS Manual Initiation One or more SFAS Functions with one channel inoperable.	1. Containment Spray (CS) Pump 1 Actuation	No	Initiate CS Pump 1	One CS and one SFAS Manual Initiation channel operable.	None	Manual initiation of CS and SFAS is not credited in the PRA model. With that being the case, the pumps for each of the CS and SFAS manual initiation buttons were chosen as a conservative surrogate.
		2. SFAS Channel 1 Actuation	No	Initiate HPI, LPI, and CCW pump 1 with associated valves.		None	
		3. CS Pump 2 Actuation	No	Initiate CS Pump 2		None	
		4. SFAS Channel 2 Actuation	No	Initiate HPI, LPI, and CCW pump 1 with associated valves.		None	
3.3.8.A.1	Emergency Diesel Generator (EDG) Loss of Power Start (LOPS) One or more Functions with one channel per bus inoperable.	Loss of Voltage (LOV) (2 channels per 4kVAC vital bus, each channel is made up of one aux relay and two TD-undervoltage relays)	Yes	Sense LOV on 4kVAC vital bus and initiate power transfer to DGs	Two-out-of-two channel actuation logic per 4kVAC vital bus. With one channel INOP on a vital bus and not tripped, the success criterion is met by either the degraded voltage function, or by the unaffected 4kVAC vital bus.	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
		Degraded Voltage (2 EDG LOPS channels per 4kVAC vital bus, each channel is made up of one aux relay and two TD-undervoltage relays)	No	Sense sustained DV on 4kVAC vital bus and initiate power transfer to DGs	Two-out-of-two channel actuation logic per 4kVAC vital bus. With one channel INOP on a vital bus and not tripped, the success criterion is met by either the degraded voltage function, or by the unaffected 4kVAC vital bus.	None.	DV function is not modeled in the PRA, therefore LOV relays are used as a conservative surrogate.
3.3.8.B.1	Emergency Diesel Generator (EDG) Loss of Power Start (LOPS) One or more Functions with two channels per bus inoperable.	See 3.3.8.A.1 functions above.	Partially	See 3.3.8.A.1 functions above.	The most limiting condition is 1 LOV channel INOP and 1 DV channel INOP. In this condition the 2-out-of-2 logic is not met for either function on the affected 4kVAC bus, and the design success criteria must be met by the unaffected 4kVAC vital bus.	For LOV, same as design success criteria. For DV, None.	Because only the LOV function is modeled, the LCO condition will fail the affected EDG. Therefore, the affected EDG is used as the surrogate for the INOP condition.
3.3.11.A.1	Steam and Feedwater Rupture Control System (SFRCS) Instrumentation One or more Functions with	Main Steam Line (MSL) Pressure – Low. (4 SFRCS channels per MSL)	Yes	Isolates MS and MFW and initiates AFW.	Two-out-of-two MSL Pressure – Low instrument channels on 1-out-of-2 logic channels of the affected MSL	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
	one channel inoperable.	Feedwater/Steam Generator Differential Pressure – High. (4 SFRCS channels per SG)	Yes	Isolates MS and MFW and initiates AFW.	Two-out-of-two Feedwater/Steam Generator Differential Pressure – High instrument channels on 1-out-of-2 logic channels of the affected SG	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
		Steam Generator Level – Low. (4 SFRCS channels per SG)	Yes	Isolates MS and MFW and initiates AFW.	Two-out-of-two Steam Generator Level – Low instrument channels on 1-out-of-2 logic channels of the affected SG	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
		Loss of RCPs. (4 SFRCS channels per RCP)	Yes	Isolates MS and MFW and initiates AFW. Trip affected RCP(s).	One-out-of-one Loss of RCPs channel per affected RCP on all four tripped RCPs	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.3.12.A.1	SFRCS Manual Initiation One or more SFRCS Functions inoperable.	AFW Pump Turbine 1 Initiation	No	Initiate AFW Pump 1	One SFRCS Manual Initiation button per function operable	None	Manual initiation of SFRCS is not credited in the PRA model. With that being the case, the SFRCS logic board events
		AFW Pump Turbine 1 and SG 1 Isolation	No	Initiate AFW Pump 1 and isolation SG 1		None	
		AFW Pump Turbine 2 Initiation	No	Initiate AFW Pump 2		None	

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
		AFW Pump Turbine 2 and SG 2 Isolation	No	Initiate AFW Pump 2 and isolation SG 2		None	were chosen as a conservative surrogate.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.3.13.A.1	SFRCS Actuation One or more channel 1 Logic Functions inoperable with all channel 2 Logic Functions OPERABLE. <u>OR</u> One or more channel 2 Logic Functions inoperable with all channel 1 Logic Functions OPERABLE.	Auxiliary Feedwater Initiation (4 SFRCS logic channels, 2 per actuation channel)	Yes	Initiate AFW	Two-out-of-two logic channels on 1-out-of-2 actuation channels: With one or more logic functions INOP on Actuation Channel 1, the success criteria is met by the unaffected Actuation Channel 2. <u>OR</u> With one or more logic functions INOP on Actuation Channel 2 actuation , the success criteria is met by the unaffected Actuation Channel 1.	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program
		Auxiliary Feedwater and Main Steam Valve Control (4 SFRCS logic channels, 2 per actuation channel)	Yes	Align AFW to provide feedwater to unaffected SG			
		Main Steam Line Isolation (4 SFRCS logic channels, 2 per actuation channel)	Yes	Isolate MSL via MSIVs and TSVs			
		Main Feedwater Isolation (4 SFRCS logic channels, 2 per actuation channel)	Yes	Isolate MFW via MFSVs, MFCVs, and SFCVs			

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.4.9.C.1	Pressurizer Capacity of essential pressurizer heaters less than limit.	Two pressurizer essential heater banks.	No	Pressure control, pressure transient mitigation, and subcooling margin	Minimum required pressurizer capacity of 85 kW ensures RCS pressure can be maintained.	None	Surrogates used for components are mapped based on their effect on the function. For the RICT, the impact has been mapped to an increase in the likelihood of a plant trip due to degraded pressure control. See note 5 for details
3.5.2.A.1	ECCS (Emergency Core Cooling System) – Operating One low pressure injection (LPI) subsystem inoperable.	Two ECCS trains of LPI.	Yes	Limit peak cladding temperature within acceptable limits in the event of a LOCA, REA, SGTR, or MSLB and long-term core cooling in recirculation mode	One LPI train operable	Same as design criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.5.2.B.1	ECCS (Emergency Core Cooling System) – Operating One or more trains inoperable for reasons other than 3.5.2.A.	Two ECCS trains. (HPI, LPI, Cont. Sump, BWST, & Valves)	Yes	Limit peak cladding temperature within acceptable limits in the event of a LOCA, REA, SGTR, or MSLB and long-term core cooling in recirculation mode	One of two ECCS trains with operable LPI, HPI, BWST, injection and recirculation pathways.	Same as design criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.6.2.C.3	Containment Air Locks One or more containment air locks inoperable for reasons other than 3.6.2.A or 3.6.2.B.	Two containment air locks.	Isolation function not modeled.	To limit fission product release during and following postulated Design Basis Accident (DBA).	Closure of a single door on each of two air locks.	None for TS containment isolation function.	See Note 3. The airlocks are not modeled for isolation so a latent containment isolation failure will be used as a conservative surrogate for the RICT calculation.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.6.3.A.1	Containment Isolation Valves ---------NOTE-------- Only applicable to penetration flow paths with only two containment isolation valve. ------------------------- One or more penetration flow paths with one containment isolation valve inoperable for reasons other than 3.6.3.D or 3.6.3.E	Containment isolation valves.	Partial	To limit fission product release during and following postulated Design Basis Accident (DBA).	One containment isolation valve closed per penetration.	Same as design success criteria.	Not all primary containment isolation valves are modeled. For valves that are not modeled, a surrogate of a latent containment failure is chosen. See Note 4 for details

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.6.3.C.1	Containment Isolation Valves --------NOTE------- Only applicable to penetration flow paths with only one containment isolation valve. ------------------------ One or more penetration flow paths with one containment isolation valve inoperable.	Containment isolation valves.	Partial	To limit fission product release during and following postulated Design Basis Accident (DBA).	One containment isolation valve closed per penetration.	Same as design success criteria.	Not all primary containment isolation valves are modeled. For valves that are not modeled, a surrogate of a latent containment failure is chosen. See Note 4 for details.
3.6.6.A.1	Containment Spray and Air Cooling Systems One containment spray train inoperable.	Two containment spray trains and two containment air cooling (CAC) trains with a swing CAC.	Yes	Maintain containment peak pressure and temperature below design limits.	One containment air cooling train and one containment spray train	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.6.6.C.1	Containment Spray and Air Cooling Systems One required containment air cooling train inoperable.	Two containment spray trains and two containment air cooling trains with a swing CAC.	Yes	Maintain containment peak pressure and temperature below design limits.	One containment air cooling train and one containment spray train	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.6.6.D (Required Actions D.1 and D.2)	Containment Spray and Air Cooling Systems One containment spray train and one required containment air cooling train inoperable.	Two containment spray trains and two containment air cooling trains with a swing CAC.	Yes	Maintain containment peak pressure and temperature below design limits.	One containment air cooling train and one containment spray train	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.6.6.E.1	Containment Spray and Air Cooling Systems Two required containment air cooling train inoperable.	Two containment spray trains and two containment air cooling trains with a swing CAC.	Yes	Maintain containment peak pressure and temperature below design limits.	One containment air cooling train and one containment spray train	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.7.2.A.1	Main Steam Isolation Valves (MSIVs) One MSIV inoperable in MODE 1.	Two MSIVs, one per SG.	Yes	Isolate affected SG following a secondary line break (SLB) requiring SG isolation.	One or both MSIVs close to prevent an uncontrolled blow down of both SGs. With one MSIV INOP, the success criteria is met by the operable MSIV and the turbine block valves to prevent uncontrolled blowdown of both SGs.	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.7.5.A.1	Emergency Feedwater (EFW) One Auxiliary Feedwater (AFW) train inoperable due to one inoperable steam supply. <u>OR</u> One AFW train inoperable in MODE 3 following refueling.	Three EFW trains, which include two AFW trains and the Motor Driven Feedwater Pump (MDFP) train.	Yes	Supply water to SGs to remove decay heat and other residual heat following a MSLB and/or loss of main feedwater.	Any single EFW train operable.	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.7.5.B.1	EFW One EFW train inoperable for reasons other than 3.7.5.A in MODE 1, 2, or 3.	Three EFW trains, which include two AFW trains and the MDFP train.	Yes	Supply water to the SGs to remove decay heat and other residual heat following a MSLB and/or loss of main feedwater.	Any single EFW train operable.	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.7.7.A.1	Component Cooling Water (CCW) System One CCW loop inoperable.	Two CCW loops.	Yes	Cooling for continued operation of safety-related equipment	One CCW loop operable.	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.7.8.A.1	Service Water System (SWS) One SWS loop inoperable.	Two SWS loops.	Yes	Cooling for continued operation of safety-related equipment	One SWS loop operable.	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.8.1.A.3	AC Sources – Operating One offsite circuit inoperable.	Two switchyard feeds, two startup transformers, two offsite power buses, and associated breakers	Yes	Source of power to safety-related systems	With one offsite circuit INOP, the design success criteria is met by 1 of 2 operable EDGs OR 1 operable offsite power circuit.	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.1.B.4	AC Sources – Operating One EDG inoperable.	Two EDGs and their support systems.	Yes	Source of power to safety-related systems	With one EDG INOP, the design success criteria is met by the operable EDG OR 1 of 2 operable offsite power circuit.	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.8.1.C.2	AC Sources – Operating Two offsite circuits inoperable.	See 3.8.1.A.3	Yes	Source of power to safety-related systems	With two offsite circuits INOP, the design success criteria is met by 1 of 2 operable EDGs.	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.8.1.D	AC Sources – Operating One offsite circuit inoperable. <u>AND</u> One EDG inoperable.	See 3.8.1.A.3 and 3.8.1.B.4	Yes	Source of power to safety-related systems	With one offsite circuit and one EDG inoperable, the design success criteria is met by the operable EDG OR the operable offsite power circuit.	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.1.G.1	AC Sources – Operating One or more trains with one load sequencer inoperable.	Four load sequencers, two per 4kVAC vital bus	Yes	Following EDG start, sequences vital loads on the 4kVAC Vital bus.	With one load sequencer INOP per train, the operable load sequencers meet the design success criteria. (1-out-of-2 logic)	Same as design success criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.8.4.A.3	DC Sources – Operating One or two required battery charges on one train inoperable.	Four battery chargers (One per battery, two per train)	Yes	Provide DC power to Train 1 and 2 loads during normal and abnormal operation	With one or two required battery chargers INOP, the design success criterion is met by the two available battery chargers, one per battery on the unaffected train.	Same as success design criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.8.4.B.1	DC Sources – Operating One DC electrical power source inoperable for reasons other than 3.8.4.A	The Train 1 and 2 125/250 VDC buses, breakers, instrumentation, and supports.	Yes	Provide DC power to Train 1 and 2 loads during normal and abnormal operation	With one DC subsystem INOP, the success criteria is met by the available DC subsystem.	Same as success design criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.8.7.A.1	Inverters – Operating One inverter inoperable.	4 inverters (One per 120 VAC vital bus)	Yes	Provide AC instrumentation power to vital buses	With one inverter INOP, the success criteria is met by the redundant inverter for the affected train. One inverter in each train meets the design success criteria.	Same as success design criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1: In-scope TS/LCO Conditions to Corresponding PRA Functions

DB TS	TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.9.A.1	Distribution Systems – Operating One or more AC electrical power distribution subsystems inoperable.	Train 1 and 2 AC electrical power distribution subsystems, consisting of 4160V and 480V AC essential buses.	Yes	Provide AC power to the associated essential buses	With one or more AC electrical power distribution subsystem(s) INOP, the success criteria is met by the unaffected train of AC electrical power distribution subsystem(s).	Same as success design criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.8.9.B.1	Distribution Systems – Operating One or more AC vital buses inoperable.	Train 1 and 2 120V AC vital bus distribution subsystems.	Yes	Provide AC power to the associated vital buses	With one AC vital bus INOP, the success criteria is met by the redundant AC vital bus for the affected train. One AC vital bus in each train meets the design success criteria.	Same as success design criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.
3.8.9.C.1	Distribution Systems – Operating One DC electrical power distribution subsystem inoperable.	Train 1 and 2 250/125V DC bus distribution subsystems.	Yes	Provide DC power to the associated essential buses	With one DC electrical power distribution subsystem INOP, the success criteria is met by the unaffected subsystem.	Same as success design criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRMP for the RICT program.

Table E1-1 Notes:

1. Individual RPS instrumentation channels are not modeled in the PRA but rather a single point estimate event representing mechanical failures of the RPS or control rods. An RPS failure model based on the Davis-Besse Model in NUREG-CR/5500 Volume 11 model (Reference 4) will be used for the configuration risk monitor program (CRMP). For the RICT calculation, the Reference 4 model cutsets were reviewed and a probability of failure of each of the four channels was developed. A simplified RPS model using the four sub-system failure events using the two-out-of-four logic (modeled as three-out-of-four failures) will be used. This simplified RPS model generates the exact probability of the NUREG-CR/5500 model for no test and maintenance and for one channel INOP. For any channels with a function considered INOP, the associated channel event was failed in the simplified RPS model to calculate the RICT. This simplified RPS model provides conservative results consistent with NUREG-CR/5500 for one train channel INOP, and more conservative, with two channels INOP due to the complete failure of the channel versus the individual function. This simplified RPS model is used to calculate the LAR estimated RICT values. The DBNPS CRMP model will be updated to use this simplified RPS model prior to RICT program implementation. This is not a model upgrade, but only a model maintenance item and does not introduce any new PRA methods.

2. SFAS actuates multiple different systems based on Incident Level of initiation, which could occur in a non-sequential order. SFAS Level 1 includes isolation of the Containment Purge and Exhaust, Containment Sampling, and Control Room Normal Ventilation Systems as well as actuation of Station Emergency Ventilation System. SFAS Level 2 includes initiation of HPI, the CCW system, the SWS system, the CAC coolers, and the EDGs. Various containment isolation valves are closed and the CS valves are opened, but the CS pumps are not started. SFAS Level 3 initiates LPI and additional containment isolation valves to close. SFAS Level 4 initiates the CS system and additional containment isolation valves to close. SFAS Level 5 enables the permissive for manual transfer of LPI pumps suction to the containment emergency sump for recirculation (Reference 6).

3. The primary containment air locks are not explicitly modeled in the DBNPS PRA for TS function to isolate containment and limit releases. It follows that there is no explicit PRA success criterion. However, the LCO condition will be modeled using the liner degradation latent containment isolation failure as a conservative surrogate in the PRA. An industry pre-existing large containment failure event probability is used as a basis to develop a conservative surrogate. This event was derived by the Pacific Northwest National Laboratory for the NRC (see EPRI Risk Impact Assessment of Extended Integrate Leak Rate Test Intervals, TR-1009325 plus the use of NUREG-1493) [References 7 and 8]. This generic value is an industry accepted estimate that addresses the operating experience-based probability of containment release pathways being larger than “small”. Because the containment hatch doors have no dependencies, for the LCO condition, it is appropriate to increase the failure probability of the surrogate event in the CMRP program (versus setting to logical True) for the RICT calculation. This added probability represents the likelihood of failure of the redundant operable door. A bounding individual door failure probability was derived by taking the square root of the latent containment isolation failure probability.

4. Where containment isolation SSCs are modeled consistent with the TS scope, unavailability can be directly included in the CRMP for the RICT program. Unmodeled PCIVs were screened in the PRA from LERF consideration based on PCIVs being smaller than 3 inches

or if the PCIV isolates a closed system inside containment. However, a conservative assessment using a surrogate latent containment isolation failure will be used to address individual unmodeled containment isolation valve unavailability (See basis for event in Table E1-1 Note 3). Although very conservative, screened penetrations shall be assessed with this surrogate. This approach is conservative because unmodeled containment isolation SSCs have been determined to not contribute to LERF.

5. The unavailability of one required group of pressurizer heaters would not have any significant impact on plant transient response so there is no quantifiable impact to CDF or LERF. Degraded pressurizer heater capability is supplemented by the availability of the remaining heaters for plant pressure control, and the availability of plant procedures which provide plant shutdown and cooldown guidance with pressurizer heaters. If the available heaters are sufficient to maintain RCS pressure control, normal plant operations can continue. For the RICT, the impact has been mapped to an increase in the likelihood of a plant trip (factor of 10) due to degraded pressure control.

RICTs were calculated for both trains when applicable and the most limiting RICT is specified in Table E1-2. Following implementation of the RICT Program, the actual RICT values will be calculated using the actual plant configuration and the current revision of the PRA model representing the as-built, as-operated condition of the plant, as required by NEI 06-09-A, Revision 0 (reference 2) and the NRC Final Safety Evaluation (reference 1).

RICTs are based on the internal events PRA (including internal flooding), internal fire PRA, and seismic penalty model calculations for CDF and LERF. RICTs calculated to be greater than 30 days are capped at 30 days based on NEI 06-09-A, Revision 0. RICTs not capped at 30 days are rounded to nearest hundredth of a day.

Per NEI 06-09-A, Revision 0, for cases where the total CDF or LERF is greater than 1E-03/yr or 1E-04/yr, respectively, the RICT will not be voluntarily entered.

Table E1-2: In-Scope TS/LCO Conditions RICT Estimate

Tech Spec	LCO Condition	RICT Estimate
3.3.1.B.1	Reactor Protection System (RPS) Instrumentation - Two channels inoperable	20.90 days[1]
3.3.5.A.1	Safety Features Actuation System (SFAS) Instrumentation - One or more Parameters with one channel inoperable.	30.00 days
3.3.6.A.1	SFAS Manual Initiation - One or more SFAS Functions with one channel inoperable.	9.56 days
3.3.8.A.1	Emergency Diesel Generator (EDG) Loss of Power Start (LOPS) - One or more Functions with one channel per bus inoperable.	30.00 days
3.3.8.B.1	Emergency Diesel Generator (EDG) Loss of Power Start (LOPS) - One or more Functions with two channels per bus inoperable.	30.00 days
3.3.11.A.1	Steam and Feedwater Rupture Control System (SFRCS) Instrumentation - One or more Functions with one channel inoperable.	30.00 days
3.3.12.A.1	SFRCS Manual Initiation - One or more SFRCS Functions inoperable.	30.00 days
3.3.13.A.1	SFRCS Actuation One or more channel 1 Logic Functions inoperable with all channel 2 Logic Functions OPERABLE. OR One or more channel 2 Logic Functions inoperable with all channel 1 Logic Functions OPERABLE.	30.00 days
3.4.9.C.1	Pressurizer - Capacity of essential pressurizer heaters less than limit.	30.00 days
3.5.2.A.1	ECCS (Emergency Core Cooling System) – Operating - One low pressure injection (LPI) subsystem inoperable.	15.67 days
3.5.2.B.1	ECCS (Emergency Core Cooling System) – Operating - One or	30.00 days

Table E1-2: In-Scope TS/LCO Conditions RICT Estimate

Tech Spec	LCO Condition	RICT Estimate
	more trains inoperable for reasons other than 3.5.2.A.	
3.6.2.C.3	Containment Air Locks - One or more containment air locks inoperable for reasons other than 3.6.2.A or 3.6.2.B.	30.00 days
3.6.3.A.1	Containment Isolation Valves ---------NOTE-------- Only applicable to penetration flow paths with only two containment isolation valve. ------------------------ One or more penetration flow paths with one containment isolation valve inoperable for reasons other than 3.6.3.D or 3.6.3.E	27.57 days
3.6.3.C.1	Containment Isolation Valves ---------NOTE-------- Only applicable to penetration flow paths with only one containment isolation valve. ------------------------ One or more penetration flow paths with one containment isolation valve inoperable.	30.00 days
3.6.6.A.1	Containment Spray and Air Cooling Systems - One containment spray train inoperable.	30.00 days
3.6.6.C.1	Containment Spray and Air Cooling Systems - One required containment air cooling train inoperable.	30.00 days
3.6.6.D[2]	Containment Spray and Air Cooling Systems - One containment spray train and one required containment air cooling train inoperable.	30.00 days
3.6.6.E.1	Containment Spray and Air Cooling Systems - Two required containment air cooling train inoperable.	30.00 days
3.7.2.A.1	Main Steam Isolation Valves (MSIVs) - One MSIV inoperable in MODE 1.	30.00 days
3.7.5.A.1	Emergency Feedwater (EFW) One Auxiliary Feedwater (AFW) train inoperable due to one inoperable steam supply.	30.00 days
3.7.5.B.1	EFW - One EFW train inoperable for reasons other than 3.7.5.A in MODE 1, 2, or 3.	30.00 days[3]
3.7.7.A.1	Component Cooling Water (CCW) System - One CCW loop inoperable.	13.69 days

Table E1-2: In-Scope TS/LCO Conditions RICT Estimate

Tech Spec	LCO Condition	RICT Estimate
3.7.8.A.1	Service Water System (SWS) - One SWS loop inoperable.	12.80 days[4]
3.8.1.A.3	AC Sources – Operating - One offsite circuit inoperable.	30.00 days
3.8.1.B.4	AC Sources – Operating - One EDG inoperable.	30.00 days
3.8.1.C.2	AC Sources – Operating - Two offsite circuits inoperable.	30.00 days
3.8.1.D[5]	AC Sources – Operating - One offsite circuit inoperable and one EDG inoperable	17.22 days
3.8.1.G.1	AC Sources – Operating - One or more trains with one load sequencer inoperable.	30.00 days
3.8.4.A.3	DC Sources – Operating - One or two required battery charges on one train inoperable.	20.38 days
3.8.4.B.1	DC Sources – Operating - One DC electrical power source inoperable for reasons other than 3.8.4.A	1.08 days
3.8.7.A.1	Inverters – Operating - One inverter inoperable.	30.00 days
3.8.9.A.1	Distribution Systems – Operating - One or more AC electrical power distribution subsystems inoperable.	3.67 days
3.8.9.B.1	Distribution Systems – Operating - One or more AC vital buses inoperable.	30.00 days
3.8.9.C.1	Distribution Systems – Operating - One DC electrical power distribution subsystem inoperable.	16.40 days

Table E1-2 Notes:

1. RICT was calculated for all cases of RPS instrumentation failure (one, two, three, and four channels of RPS inoperable). The RICT presented is the case for the TS 3.3.1.B.1 LCO condition of two channels inoperable. The most limiting case with four RPS instrumentation channels inoperable was calculated to 0.03 days.
2. Includes required actions D.1 and D.2.
3. Quantification case was not created as it would be identical to 3.7.5.A.1.
4. The RICT presented is the case for the TS 3.7.8.A.1 LCO condition of one running and one standby SSW pump unavailable. The worst case scenario with two running SSW pumps unavailable was calculated to 3.73 days, but does not properly represent the LCO condition.
5. Includes required actions D.1 and D.2.

2.0 Additional Justification for Specific Actions

This section contains the additional technical justification for the list of Required Actions from Table 1, "Conditions Requiring Additional Technical Justification", of TSTF-505, Revision 2.

Additional justification for each of the identified DBNPS TS is provided below:

Table E1-3: TSTF-505 Rev 2 Table 1 Technical Specifications (TS) that Require Additional Justification

TSTF-505 TS Condition	DBNPS TS Condition	DBNPS TS Condition Description	Additional Justification
3.3.8.A and 3.3.8.B	3.3.8.A and 3.3.8.B	TS 3.3.8: Emergency Diesel Generator (EDG) Loss of Power Start (LOPS) LCO: Two channels of Loss of Voltage Function and two channels of Degraded Voltage Function EDG LOPS instrumentation per bus shall be OPERABLE. Condition A: One or more Functions with one channel per bus inoperable. Condition B: One or more Functions with two channels per bus inoperable.	TSTF-505 changes are incorporated. Undervoltage protection will generate a LOPS in the event a loss of voltage or degraded voltage condition occurs in the switchyard. There are two LOPS Functions for each 4.16 kV vital bus. Each function has two channels arranged in two-out-of-two logic for each essential bus. If only the channels of one essential bus are affected, then single failure protection for the required function may be lost; however, the function itself has not been lost since the channels of the other essential bus are unaffected. Because DBNPS TS 3.3.8 Conditions A and B govern one or two channels per bus inoperable for one or more functions, a loss of function could result if one or more channels for that function in both buses are inoperable. Therefore, Vistra proposes a Note be inserted prior to the RICT statement preventing application of a RICT if a loss of function occurs (see Attachment 2). The associated TS 3.3.8 Bases are also modified to include discussion of

Table E1-3: TSTF-505 Rev 2 Table 1 Technical Specifications (TS) that Require Additional Justification

TSTF-505 TS Condition	DBNPS TS Condition	DBNPS TS Condition Description	Additional Justification
			configurations that result in a loss of safety function.
3.3.12.B	3.3.12.A	<u>TS 3.3.12: Steam and Feedwater Rupture Control System (SFRCS) Manual Initiation</u> LCO: One manual initiation push button for each of the following SFRCS Functions shall be OPERABLE: a. Auxiliary Feedwater Pump Turbine 1 Initiation; b. Auxiliary Feedwater Pump Turbine 2 Initiation; c. Auxiliary Feedwater Pump Turbine 1 Initiation and Steam Generator 1 Isolation; and d. Auxiliary Feedwater Pump Turbine 2 Initiation and Steam Generator 2 Isolation. Condition A: One or more SFRCS Functions inoperable.	TSTF-505 changes are incorporated. The SFRCS manual initiation capability provides the operator with the capability to actuate SFRCS Functions from the control room in the absence of any other initiation condition. Manually actuated Functions include Auxiliary Feedwater Pump Turbine (AFPT) 1 Initiation; AFPT 2 Initiation; AFPT 1 Initiation and Steam Generator (SG) 1 Isolation; and AFPT 2 Initiation and SG 2 Isolation. SFRCS Functions credited in the safety analysis are automatic. However, the manual initiation Functions are required by design as backups to the automatic trip Functions and allow operators to initiate AFPT and actuate SG isolation whenever these Functions are needed. The SFRCS manual initiation Functions are retained for the overall redundancy and diversity of the SFRCS as required by the NRC. With one or more SFRCS functions with the required manual initiation push button inoperable in one actuation train, the redundant train may still be actuated via the operable manual initiation push button. Additionally, loss of the manual initiation capability does not prevent the automatic actuation features of SFRCS. Because the accident analyses

Table E1-3: TSTF-505 Rev 2 Table 1 Technical Specifications (TS) that Require Additional Justification

TSTF-505 TS Condition	DBNPS TS Condition	DBNPS TS Condition Description	Additional Justification
			assume SFRCS will actuate automatically and automatic features remain available, the safety functions will continue to be met. Manual actuation is an operator aid and is not relied upon in the accident analyses. Therefore, this Condition meets the requirements for inclusion in the RICT Program.

Table E1-3: TSTF-505 Rev 2 Table 1 Technical Specifications (TS) that Require Additional Justification

TSTF-505 TS Condition	DBNPS TS Condition	DBNPS TS Condition Description	Additional Justification
3.4.9.C	3.4.9.C	TS 3.4.9: Pressurizer LCO: The pressurizer shall be OPERABLE with: a. Pressurizer water levels 228 inches; and b. A minimum of 85 kW of essential pressurizer heaters OPERABLE. Condition C: Capacity of essential pressurizer heaters less than limit.	TSTF-505 changes are incorporated. The unavailability of one required group of pressurizer heaters would not have any significant impact on plant transient response so there is no quantifiable impact to CDF or LERF. Degraded pressurizer heater capability is supplemented by the availability of the remaining heaters for plant pressure control, and the availability of plant procedures which provide plant shutdown and cooldown guidance with pressurizer heaters. If the available heaters are sufficient to maintain RCS pressure control, normal plant operations can continue. The pressurizer heaters will be evaluated for the RICT Program by a bounding assessment as permitted by NEI 06-09-A. For the RICT, the impact has been mapped to an increase in the likelihood of a plant trip (factor of 10) due to degraded pressure control. This surrogate is consistent with recently approved TSTF-505, Revision 2 applications for plants similar in design to DBNPS. Therefore, this Condition meets the requirements for inclusion in the RICT Program.
3.5.2.B	3.5.2.B	TS 3.5.2: ECCS – Operating LCO: Two ECCS trains shall be OPERABLE. Condition B:	TSTF-505 changes are incorporated. The function of the ECCS is to provide core cooling to ensure that the reactor core is protected after an accident. Two redundant, 100% capacity trains are provided.

Table E1-3: TSTF-505 Rev 2 Table 1 Technical Specifications (TS) that Require Additional Justification

TSTF-505 TS Condition	DBNPS TS Condition	DBNPS TS Condition Description	Additional Justification
		One or more trains inoperable for reasons other than Condition A.	Condition A involves one low pressure injection (LPI) Subsystem inoperable. Application of the RICT to TS 3.5.2 Required Action B.1 is acceptable because Condition D prevents continued operation when 100% of the flow equivalent to a single operable ECCS train is not available. In such cases, LCO 3.0.3 must be entered immediately, ensuring the unit is placed in a safe condition when a loss of safety function exists. Therefore, this Condition meets the requirements for inclusion in the RICT Program.
3.6.2.C	3.6.2.C	<u>TS 3.6.2: Containment Air Locks</u> LCO: Two containment air locks shall be OPERABLE. Condition C: One or more containment air locks inoperable for reasons other than Condition A or B.	TSTF-505 changes are incorporated. DBNPS air locks form part of the DBNPS pressure boundary and provide a means for personnel access during all modes of operation. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Table E1-3: TSTF-505 Rev 2 Table 1 Technical Specifications (TS) that Require Additional Justification

TSTF-505 TS Condition	DBNPS TS Condition	DBNPS TS Condition Description	Additional Justification
			Application of the RICT to TS 3.6.2 Required Action C.3 is acceptable because when both doors in one or more air locks are inoperable, DBNPS leakage must be assessed in accordance with Required Action C.1. If leakage limits are exceeded, LCO 3.6.1, "Containment," requires restoration within 1 hour or a unit shutdown must be performed in accordance with LCO 3.6.1 Required Actions B.1 and B.2, ensuring the unit is placed is a safe condition. Therefore, this Condition meets the requirements for inclusion in the RICT Program.
3.6.6	3.6.6	<u>TS 3.6.6: Containment Spray and Air Cooling Systems</u> LCO: Two containment spray trains and two containment air cooling trains shall be OPERABLE. Condition A: One containment spray train inoperable. Condition C: One required containment air cooling train inoperable.	TSTF-505 changes are incorporated. The Containment Spray and Containment Air Cooling (CAC) Systems provide containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. The SSCs associated with these functions are explicitly modeled in the DBNPS PRA. The PRA success criteria for Containment Spray and CAC are 1 of 2 trains and 1 of 2 trains with a swing CAC respectively, which is the same as the design success criteria for the system. The function covered by DBNPS LCO 3.6.6 is to

Table E1-3: TSTF-505 Rev 2 Table 1 Technical Specifications (TS) that Require Additional Justification

TSTF-505 TS Condition	DBNPS TS Condition	DBNPS TS Condition Description	Additional Justification
		Condition D: One containment spray train and one required containment air cooling train inoperable. Condition E: Two required containment air cooling trains inoperable.	maintain containment peak pressure and temperature below design limits. The SSCs for Containment Spray and CAC are modeled in the DBNPS PRA consistent with the TS scope and can be directly evaluated. Since the containment spray SSCs are adequately modeled in the DBNPS PRA and a RICT can be calculated for the conditions, DBNPS LCO 3.6.6 Actions A.1, C.1, D.1, D.2, and E.1 meet the listed requirements for inclusion in the RICT Program.
3.7.2.A	3.7.2.A	TS 3.7.2: Main Steam Isolation Valves (MSIVs) LCO: Two MSIVs shall be OPERABLE. Condition A: One MSIV inoperable in MODE 1.	TSTF-505 changes are incorporated. The MSIVs isolate steam flow from the secondary side of the steam generators following a main steam line break (MSLB) or feedwater line break. MSIV closure terminates flow from the unaffected (intact) steam generator. The turbine stop valves (TSVs) also provide a means for main steam isolation in the event of an MSLB. Closure of the TSVs ensures that both steam generators do not blow down following an MSLB in conjunction with the MSIV associated with the unaffected steam generator failing to close. The design of the MSIVs precludes the blowdown of more than one SG, assuming a single active component failure (e.g., the failure of one MSIV to close on demand). With one MSIV inoperable

Table E1-3: TSTF-505 Rev 2 Table 1 Technical Specifications (TS) that Require Additional Justification

TSTF-505 TS Condition	DBNPS TS Condition	DBNPS TS Condition Description	Additional Justification
			in Condition A, the steam line isolation function of TS 3.7.2 is met by the remaining OPERABLE MSIV on the other SG, the TSVs, and the non-return valves to prevent blowdown of more than one SG. As a result, the loss of a single MSIV will not result in a loss of the steam line isolation function. Therefore, this Condition meets the requirements for inclusion in the RICT Program.

3.0 Electrical Asymmetry

At times, there are large differences in RICTs between alternate trains of the same electrical system. This is mostly clearly presented in the distribution system LCOs (3.8.9) and is demonstrated to the greatest extent by the DC distribution system required action (3.8.9.C.1). The LCO is applicable when there is one DC electrical power distribution subsystem inoperable and is covered by 250/125V buses. The calculated RICT durations for both 250/125V buses is shown below:

EBDZ086F: MCC 1 LOCAL FAULTS - 240.7 days

EBDZ087F: MCC 2 LOCAL FAULTS - 16.4 days

The large difference in calculated RICT durations is due to many complex and overarching factors within the model, which include but are not limited to plant configuration, fire and internal flooding spatial considerations, and differing electrical dependencies. These differences will be monitored for and addressed as part of the RICT program implementation.

4.0 Modeling of FLEX

Vistra intends to credit the PRA modeling of FLEX equipment in the RICT application. Therefore, this section provides background information, a summary of FLEX strategies that are modeled in the PRA, the modeling methodology, and the results of sensitivity analyses.

Background

The Diverse and Flexible Coping Strategies (FLEX) system is a combination of pre-planned approaches to establish an indefinite coping capability during which key safety functions (core cooling, containment integrity, and spent fuel pool cooling) are maintained during an extended loss of all AC power concurrent with a loss of normal access to the ultimate heat sink, as required by NRC Order EA-12-049 (Reference 9). The DBNPS FLEX program is based upon NEI 12-06 (Reference 10) and is implemented through Vistra programs and procedures. The station's ability to successfully execute FLEX strategies, such as deployment of portable equipment from its storage location to its designated staging point, have been demonstrated and documented within the EFW and FLEX PRA notebook (Reference 11).

Modeling of FLEX Equipment and Actions

FLEX equipment is credited in the DBNPS internal events, internal flooding, and internal fire PRA models. The FLEX equipment modeled in the DBNPS PRA includes:

- FLEX Turbine Marine 480V Generators (FX-K1P and FX-K1A), only one of the generators is credited that can be manually connected to the 480V MCC to supply power of the emergency feedwater facility (EFWF) after a loss of offsite power.
- FLEX RCS Charging Pumps (P296-1 and P296-2), two positive displacement pumps permanently staged in the auxiliary building which must be connected to the reactor coolant system (RCS) and electrical power via temporary hoses and cables respectively.
- Alternate Low Pressure FLEX Emergency Feedwater (EFW) 'N' Pump(s) (FX-P1P). Two, but only one of which is modeled in the PRA, trailer mounted pump(s) that could be used to feed a depressurized steam generator.

The FLEX 480V generators must be locally and manually connected and started after a loss of offsite power accident. One generator is credited and must be aligned and able to provide power to either the EFWF and/or the FLEX RCS charging pumps when the regular EDGs are unavailable.

At least one FLEX RCS charging pump must supply water to the RCS within 8 hours of an reactor coolant pump (RCP) seal loss of coolant accident (LOCA) or a stuck open PORV/pressurizer safety valve when other injection sources are unavailable.

The Alternate Low Pressure FLEX Emergency Feedwater Pump is only credited if the EFW pump successfully starts and runs for at least an hour, as decay heat would be reduced enough to allow time to connect the low pressure pump prior to core damage. Only one of the Alternate Low Pressure FLEX Emergency Feedwater Pumps is credited in the PRA. This is due to the fact there is likely not to be sufficient time to retrieve and connect FX-P1A, which is located outside of the EFWF.

Methodology

The FLEX strategies discussed above were incorporated into the PRA models as part of the PRA maintenance process during revision 5.

Following the peer reviewed pre-initiating human reliability analysis (HRA) event screening methodology employed for non-FLEX SSCs in the internal events model, no pre-initiating HRA events unique to the FLEX equipment were developed within the EFW fault tree model. The top gates that encompass all EFW and FLEX failures are EF0001, EF0001-MCR, EF1000, EF1000-L, and FLEX as stated in the EFW and FLEX PRA notebook (Reference 11).

FLEX operator actions modeled for implementation of the above strategies are evaluated using technical approaches consistent with the endorsed ASME/ANS RA-Sa-2009 PRA Standard (Reference 3).

The failure to start and failure to run data for the FLEX equipment was developed using the generic values in PWROG-18042-P, Rev. 1 (Reference 12).

Sensitivity Analysis

Sensitivity analyses were performed to measure the impact of FLEX component reliability data and human error probabilities on selected individual and combination LCO RICT durations. For the analysis, FLEX equipment probabilities, HEPs, and CCF events were set to their 95^{th} percentile values. For the sensitivities, no RICTs changed by an amount greater or equal to 5% of the base RICT duration, which is the adopted threshold for investigating an area of uncertainty further as a key source of uncertainty toward the RICT program. Therefore, the FLEX strategy and related data are not considered key sources of uncertainty toward the RICT program.

5.0 REFERENCES

1. Letter from the U.S. NRC to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)", dated May 17, 2007 (ADAMS Accession No. ML071200238)

2. NEI Topical Report NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines", Revision 0, dated October 2012 (ADAMS Accession No. ML12286A322)

3. ASME Standard ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", dated February 2, 2009

4. U.S. NRC, NUREG/CR-5500, Volume 3, "Reliability Study: General Electric Reactor Protection System, 1984-1995", dated February 1999

5. U.S. Nuclear Regulatory Commission, Davis-Besse Nuclear Power Station, Unit No. 1 Renewed Facility Operating License – Technical Specifications, Amendment No. 306. ADAMS Accession No: ML053110490.

6. Davis-Besse Nuclear Power Station, Unit No. 1, Technical Specifications Basis, Revision 35, ADAMS Accession No: ML24269A064.

7. Electric Power Research Institute (EPRI) Technical Report (TR)-1009325, Revision 2, August 2007, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated June 25, 2008 (ADAMS Accession No. ML081140105).

8. U.S. NRC, "Performance-Based Containment Leak-Test Program," NUREG-1493, September 1995.

9. U.S. NRC Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012 (ML12054A735)

10. Nuclear Energy Institute (NEI), NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 0, dated August 2012 (ML12242A378)

11. PRA-DB1-AL-R07, Davis-Besse Nuclear Power Station PRA notebook 04-29: Emergency Feedwater System & FLEX Equipment, September 2024.

12. Pressurized Water Reactor Owner's Group (PWROG) report, PWROG-18042-P, "FLEX Equipment Data Collection and Analysis," Revision 1, PA-RMSC-1651 R1, dated August 2021

Enclosure 2

Davis-Besse Nuclear Power Station

License Amendment Request

Revise Technical Specifications to Adopt RISK Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b"

Information Support Consistency with Regulatory Guide 1.200, Revision 3

**Information Supporting Consistency with
Regulatory Guide 1.200, Revision 3**

1.0 Introduction

The purpose of this enclosure is to provide information on the technical adequacy of the Davis-Besse Nuclear Power Station (DBNPS) probabilistic risk assessment (PRA) internal events model, internal flooding model, seismic model, and fire model in support of the license amendment request to adopt TSTF-505 [Ref. 1].

This enclosure provides information supporting the DBNPS evaluation of the technical adequacy of the PRA models supporting the RICT program based on peer reviews and self-assessments against the relevant PRA standards as endorsed in the current applicable revision of RG 1.200, including consideration of staff clarifications of the standards.

Although DBNPS will transition to RG 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 3, going forward, the Nuclear Regulatory Commission (NRC) finds it remains acceptable to refer to RG 1.200, Revision 2 for demonstrating technical adequacy of the CNS PRA models. As stated in RG 1.200, Revision 3:

> “The staff’s previous endorsement of ASME/ANS RA-Sa-2009 in RG 1.200, Revision 2, issued March 2009, (Ref. 24 [sic]), has been modified to reflect endorsement of terms and their definitions derived from PWROG-19027-NP, Revision 2. However, all other portions of the staff’s endorsement of ASME/ANS RA-Sa-2009 from RG 1.200, Revision 2, are otherwise not changed by this revision, as provided in Appendix A to this RG.”

Hence it is acceptable to refer to RG 1.200, Revision 2, to demonstrate technical acceptability of the CNS PRA models.

> Note: The list of external hazards in Enclosure 4 of this LAR evaluates the hazards listed in Tables D-1 and D-2 in Appendix D of RG 1.200, Revision 3.

Per NEI 06-09-A [Ref.2], Capability Category II of the standards is applicable; therefore, this enclosure identifies those parts of the PRAs that conform to capability categories lower than Category II and provides a disposition for the RICT application. Consistent with RG 1.200, Section C.4.2, this enclosure identifies and provides a discussion of the resolutions of any findings and observations from the peer reviews or self-assessments.

This enclosure addresses the clarifications and qualifications found in RG 1.200 as the peer reviews and self-assessments performed included consideration of the clarifications and qualifications of RG 1.200 Revision 2.

The guidance in NEI 05-04 [Ref. 3], NEI 07-12 [Ref. 4], and NEI 12-13 [Ref. 5], “External Hazards PRA Peer Review Process Guidelines, Rev 0, August 2012,” to close PRA peer review findings was used for the DBNPS model as discussed in the remaining subsections of this enclosure.

Topical Report NEI 06-09-A [Ref. 2], as clarified by the United States Nuclear Regulatory Commission (NRC)'s final safety evaluation of this report [Ref. 6], defines the technical attributes of a PRA model and its associated Configuration Risk Management Program (CRMP) required to implement this risk-informed application. Meeting these requirements satisfies NRC Regulatory Guide (RG) 1.174 [Ref. 7] requirements for risk-informed plant-specific changes to a plant's licensing basis.

Vistra employs a multi-faceted approach to establishing and maintaining the technical adequacy and fidelity of PRA models. This approach includes both a proceduralized PRA maintenance and update process and the use of self-assessments and independent peer reviews.

Section 2 of this enclosure describes requirements related to the scope, as well as the technical adequacy of the DBNPS PRA internal events model, the DBNPS internal flooding PRA model, the DBNPS fire PRA model, and the DBNPS Seismic PRA model. Section 3 lists references used in the development of this enclosure. Note that this enclosure does not discuss risk impacts of external events. The treatment of external hazards is discussed in Enclosure 4.

All the PRA models described below have been peer reviewed, and the review and closure of all suggestion and finding-level F&Os from the peer reviews have been independently evaluated to confirm that the associated model changes did not constitute a model upgrade. Expectations regarding preparation of the peer reviews (NEI 05-04 [Ref. 3], Section 4.2) and conduct of the self-assessment by the host utility (NEI 05-04 [Ref. 3], Section 4.3), were addressed prior to conduct of any of the reviews. This included documentation by the host utility of resolution of the prior PRA peer review finding-level F&Os and preparation of the information required for the independent assessment. The documented bases for F&O and preparation of the information required for the independent assessment. The documented bases for F&O closure provided by DBNPS included a written assessment of whether the resolution constituted PRA maintenance or PRA upgrade.

2.0 Scope and Technical Adequacy of DBNPS Internal Events PRA Model

The DBNPS internal events PRA model is an at-power model (i.e., directly address plant configurations during plant modes 1 and 2 or reactor operation). The Level 1 and full Level 2 PRA models provide both the Core Damage Frequency (CDF) and the Large Early Release Frequency (LERF) figures of merit.

Topical report NEI 06-09-A [Ref. 2] requires that the PRA be reviewed to the guidance of NRC RG 1.200 [Ref. 8] for a PRA which meets Capability Category (CC) II for the supporting requirements of the ASME/ANS PRA Standard [Ref. 9]. It also requires that deviations from these CCs relative to the Risk-Informed Completion Time (RICT) program be justified and documented, as necessary.

The information provided in this section demonstrates that the DBNPS internal events model meets the expectations for the PRA scope and technical adequacy as presented in NRC RG 1.200, Revision 2 [Ref. 8].

Internal Events PRA Model

The DBNPS internal events PRA model was subject to a full scope peer review in 2000 conducted by the B&W Owners Group. Scientech performed a gap analysis in April 2008 to determine compliance with ASME PRA Standard RA-Sb-2005 and RG 1.200 Revision 1.

An independent F&O closure review was performed against Revision 2 of RG 1.200, by the Pressurized Water Reactor Owners Group (PWROG) in October 2017 to document review and closure of F&Os. Of the original 23 F&Os, 22 were closed and one remains open. The open F&O has been addressed and is in the current MOR, however it has not been formally closed via a peer review. This resulted in 20 SRs previously assessed as not met or Met at CC I are now MET (CC I-III) or Met at CC II or higher. Two F&Os were closed with an upgrade. The SRs related to the upgrades were peer reviewed later in October 2017 and found to be met at least at CC-II with no new findings.

Internal Flooding PRA Model

In July 2012 the DBNPS Internal Flooding PRA model was subject to a focused scope peer review by the PWROG. This peer review determined compliance with Part 3 of ASME/ANS PRA Standard RA-Sa-2009 and Addendum A of RG 1.200 Revision 2.

An independent F&O closure review was performed by the PWROG in October 2017 to document review and closure of F&Os. Of the original 11 F&Os, all 11 were closed. Three SRs that were not originally met are now fully met.

Seismic PRA Model

The PWROG performed a focused scope peer review of the Seismic PRA model in 2014. An independent F&O closure review was performed by the PWROG in October 2017 to document review and closure of F&Os from the 2014 review. Of the original 45 F&Os, all 45 were closed. This resulted in 22 SRs originally assessed as not met or met at CC-I are now met at least at CC II. Four F&Os were closed with an upgrade. The SRs related to the upgrades were peer reviewed later in October 2017 and found to be met at least at CC-II with no new findings.

Fire PRA Model

Westinghouse performed an independent peer review of the Fire PRA model in 2013. An independent F&O closure review was performed by the PWROG in October 2017 to document review and closure of F&Os from the 2013 review. Of the original 35 F&Os, 34 were closed and one remained open. This resulted in 12 SRs originally assessed as not met or Met at CC-I are now Met at least at CC-II or higher. One F&O was closed with an upgrade. The SR related to the upgrade was peer reviewed later in October 2017 and found to be met at CC I-III. The F&O that remained open at the end of the October 2017 review was addressed and peer reviewed as an upgrade in November 2017. All associated SRs were determined to be Met at least at CCII or higher. Five finding level F&Os were identified by the review. All five open findings have been addressed by making appropriate changes to the PRA model as identified by the F&O, but have not been formally closed by peer review. However, Vistra's disposition of these open F&Os has been accepted by the NRC during their review of the TSTF-425 LAR.

Summary

The peer reviews and F&O closure review demonstrate that the internal events, internal flooding, seismic, and fire models are of sufficient quality and level of detail to support the RICT program and have been subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

There are six open finding-level F&Os from the DBNPS peer reviews of the PRA models. These items have been addressed in the current model but have not been peer reviewed to formally close the findings.

3.0 References

1. TSTF-505, Revision 2, Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b, (ADAMS Accession No. ML18183A493)
2. NEI Topical Report NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines", Revision 0, dated October 2012 (ADAMS Accession No. ML12286A322)
3. NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard", Revision 2, dated November 2008 (ADAMS Accession No. ML083430462)
4. NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines", Revision 1, dated June 2010 (ADAMS Accession No. ML102230070)
5. NEI 12-13, "External Hazards PRA Peer Review Process Guidelines", Revision 0, dated August 2012 (ADAMS accession No. ML12240A027)
6. Letter from the NRC to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)", dated May 17, 2007 (ADAMS Accession No. ML071200238)
7. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", Revision 3, dated January 2018 (ADAMS Accession No. ML17317A256)
8. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Revision 2, dated March 2009 (ADAMS Accession No. ML090410013)
9. ASME/ANS Ra-Sb-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", 2009

Enclosure 3

Davis-Besse Nuclear Power Station

<u>License Amendment Request</u>

<u>Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b"</u>

Information Supporting Technical Adequacy of PRA models without PRA Standards Endorsed by Regulatory Guide 1.200, Revision 3

Information Supporting Technical Adequacy of PRA Models Without PRA Standards Endorsed by Regulatory Guide 1.200, Revision 3

This enclosure is not applicable to the Davis-Besse Nuclear Power Station (DBNPS) submittal. Vistra OpCo is not proposing to use any PRA models in the DBNPS Risk Informed Completion Time Program for which a PRA standard, as endorsed by the NRC in RG 1.200, Revision 3, does not exist.

Enclosure 4

Davis-Besse Nuclear Power Station

License Amendment Request

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b”

Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models

Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models

1. Introduction and Scope

Nuclear Energy Institute (NEI) Topical Report NEI 06-09-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,” Revision 0-A (Reference 1), as clarified by the NRC Final Safety Evaluation (Reference 2), requires that the license amendment request (LAR) provide a justification for exclusion of risk sources from the Probabilistic Risk Assessment (PRA) model based on their insignificance to the calculation of configuration risk, and to discuss conservative analyses applied to the configuration risk calculation. This enclosure addresses this requirement by discussing the overall generic methodology to identify and disposition such risk sources, and by providing the Davis-Besse Nuclear Power Station (DBNPS)-specific results of the application of the generic methodology and the disposition of impacts on the DBNPS Risk-Informed Completion Time (RICT) Program. Section 3 of this enclosure presents the plant-specific conservative analysis of seismic risk to DBNPS. Section 4 presents the justification for excluding analysis of external hazards from the DBNPS PRA.

NEI 06-09-A does not provide a specific list of hazards to be considered in a RICT program. However, non-mandatory Appendix 6-A of the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) RA-Sa-2009 PRA Standard (hereafter “ASME/ANS PRA Standard”), “Addenda to ASME/ANS RA-S-2008 for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications” (Reference 3) provides a guide for identification of most of the possible external events for a plant site. Additionally, NUREG-1855 Revision 1 (Reference 4) provides a discussion of hazards that should be evaluated to assess uncertainties in plant PRAs and support the risk-informed decision-making process. These hazards were reviewed for DBNPS, along with a review of information pertaining to the site region and plant design to identify the set of external events to be considered. Information from the DBNPS Updated Final Safety Analysis Report (UFSAR) (Reference 5) pertaining to the geologic, seismologic, hydrologic, and meteorological characteristics of the site region, and the current and projected industrial activities in the plant vicinity was reviewed. No new site-specific or plant-unique external hazards were identified through this review. The hazards from Appendix 6-A of the ASME/ANS PRA Standard that were considered for DBNPS are summarized in Table E4-1.

The scope of this enclosure is consideration of the hazards listed in Table E4-1 for applicability to DBNPS. Seismic events are evaluated quantitatively in Section 3, and the other listed external hazards are evaluated and screened as low risk in Section 4.

2. Technical Approach

The guidance contained in NEI 06-09-A states that all hazards that contribute significantly to incremental risk of a configuration must be quantitatively assessed in the implementation of the RICT Program. The following approach focuses on the risk implications of specific external hazards in the determination of the risk management action time (RMAT) and RICT for the Technical Specification (TS) Limiting Conditions for Operation (LCOs) selected as part of the RICT Program.

Consistent with NUREG-1855, Revision 1, external hazards may be addressed as follows:

1. Screening the hazard based on a low frequency of occurrence,
2. Conservatively assess the potential impact and including it in the decision-making, or
3. Developing a PRA model to be used in the RMAT/RICT calculation.

The overall process for addressing external hazards considers two aspects of the external hazard contribution to risk:

- The first is the contribution from the occurrence of beyond design basis conditions (e.g., winds greater than design). These beyond design basis conditions challenge the capability of the systems, structures, and components (SSCs) to maintain functionality and support safe shutdown of the plant.
- The second aspect addressed is the challenges caused by external conditions that are within the design basis, but still require some plant response to assure safe shutdown (e.g., high winds causing loss of offsite power, etc.). While the plant design basis assures that safety-related equipment necessary to respond to these challenges is protected, the occurrence of these conditions nevertheless causes a demand on these systems that in and of itself presents a risk.

2.1 <u>Hazard Screening</u>

The first step in the evaluation of the external hazard is screening based on an estimation of a conservative core damage frequency (CDF) for beyond design basis hazard conditions. An example of this type of screening is reliance on the NRC 1975 Standard Review Plan (SRP) (Reference 6) which is acknowledged in the NRC Individual Plant Examination of External Events (IPEEE) procedural guidance (Reference 7) as assuring a conservative CDF of less than 1E-6 per year for each hazard. The conservative CDF estimate is often characterized by the likelihood of the site being exposed to conditions that are beyond the design basis limits and an estimate of the conservative conditional core damage probability (CCDP) for those conditions. If the conservative CDF for the hazard can be shown to be less than 1E-6 per year, then beyond design basis challenges from the hazard can be screened and do not need to be assessed quantitatively in the RICT Program. The basis for this is as follows:

- The overall calculation of the RICT is limited to an incremental core damage probability (ICDP) of 1E-5.
- The maximum time interval allowed for the RICT is 30 days.
- If the maximum CDF contribution from a hazard is <1E-6 per year, then the maximum ICDP from the hazard is <1E-7 (1E-6/year * 30 days/365/days/year).
- Thus, the conservative ICDP contribution from the hazard is shown to be less than 1% of the permissible ICDP in the conservative time for the condition. Such a minimal contribution is not significant to the decision in computing a RICT.

The DBNPS hazard screening analysis from the IPEEE (Reference 8) has been updated to reflect current site conditions. The results are discussed in Section 4 and show that all events listed in Table E4-1 can be screened for DBNPS, except for seismic activity, internal flooding, and internal fire, all of which are already modeled in the DBNPS PRA.

While the direct CDF contribution from beyond design basis hazard conditions can be shown to be insignificant using this approach, some external hazards can cause a plant challenge even for hazard severities that are less than the design basis limit. These considerations are addressed in Section 4.

2.2 Hazard Analysis for CDF Contribution

There are two options in cases where the conservative CDF for the external hazard cannot be shown to be less than 1E-6 per year. The first option is to develop a PRA model that explicitly models the challenges created by the hazard and the role of the SSCs included in the RICT Program in mitigating those challenges. The second option for addressing an external hazard is to compute a conservative CDF contribution from the hazard. The conservative approach used to assess the seismic CDF contribution is described in Section 3.

2.3 Evaluation of Conservative Large Early Release Frequency (LERF) Contribution

The RICT Program requires addressing both core damage and large early release risk. When a comprehensive PRA does not exist, the LERF considerations can be estimated based on the relevant parts of the internal events LERF analysis. This can be done by considering the nature of the challenges induced by the hazard and relating those to the challenges considered in the internal events PRA. This can be done in a realistic manner or a conservative manner. The goal is to provide a representative or conservative estimate of conditional large early release probability (CLERP) that aligns with the conservative CDF evaluation. The incremental large early release frequency (ILERF) is then computed as:

$$ILERF_{Hazard} = ICDF_{Hazard} * CLERP_{Hazard}$$

The conservative approach used to assess the seismic LERF contribution is described in Section 3.

2.4 Risks from Hazard Challenges

Upon estimation of a conservative CDF and LERF, the analysis approach must ensure that the RICT Program calculations reflect the change in CDF and LERF caused by out-of-service equipment. As discussed in Section 3, seismic risk is the only beyond design basis hazard that could not be screened out for DBNPS. The approach to be used for the RICT Program considers that the change in risk with equipment out-of-service will not be higher than the conservative seismic CDF or LERF.

The above steps address the direct risks from damage to the facility from external hazards. While the direct CDF contribution from beyond design basis conditions can be shown to be insignificant without a full PRA, there may be risks that are related to the fact that some external hazards can cause a plant challenge even for hazard severities that are less than the design basis limit. For example, high winds, tornadoes, and seismic events below design basis levels can cause extended LOOP conditions. Additionally, depending on the site, external floods can challenge the availability of normal plant heat removal mechanisms.

The approach to be taken in this step is to identify the plant challenges caused by the occurrence of the hazard within the design basis and evaluate whether the risks associated with these events are either already considered in the existing PRA model or they are not significant to the risk. Section 3 provides the analysis of the beyond design basis seismic hazards for the DBNPS site, and Section 4 provides an analysis of the representative external hazards for DBNPS.

3. Conservative Seismic Analysis

Although a seismic PRA (SPRA) is available for DBNPS, the site has elected to utilize a conservative analysis of the potential seismic impact for inclusion in the decision-making process. The process for analyzing an unscreened external hazard without the use of a full PRA involves the following three steps:

1. Conservatively Estimate CDF
2. Evaluate Potential Risk Increases Due to Out-of-Service Equipment

3. Qualitatively Evaluate LERF Contribution

3.1 Conservatively Estimate Seismic CDF

As stated in the IPEEE (Reference 8), DBNPS was originally classified in NUREG-1407 (Reference 7) as a focused-scope plant, but the site provided justification for reclassification as a reduced-scope plant based on low seismicity. For the IPEEE seismic evaluation, emphasis was placed on conducting detailed seismic walkdowns utilizing the EPRI Seismic Margins Method (SMM). Success Path Logic Diagrams (SPLD) were developed to identify the systems that must function in order to successfully cool the reactor core following the occurrence of the review level earthquake (RLE).

Since DBNPS has elected to utilize a seismic penalty, an alternative approach was taken to conservatively estimate SCDF. This approach is based on the current DBNPS seismic hazard curves and conservatively estimate the seismic capacity of a component whose seismic failure would lead directly to core damage. The estimation of the SCDF uses the plant-level high confidence low probability of failure (HCLPF) seismic capacities and convolves the corresponding failure probabilities as a function of the seismic hazard level with the seismic hazard curves. This is a commonly used approach to conservatively estimate SCDF; see Section 10-B.9 of the ASME/ANS PRA Standard. This approach is consistent with approaches that have been used in other regulatory applications.

Based on the DBNPS foundation input response spectra (FIRS) curve, the plant-level HCLPF is 0.2g peak ground acceleration (PGA). IPEEE studies previously estimated the HCLPF as 0.26g. Because the FIRS-derived HCLPF is more conservative, it is used as the HCLPF for this analysis.

Using the DBNPS HCLPF of 0.2g, calculations were performed for frequencies of 1 Hz, 5 Hz, 10 Hz, and 100 Hz. The DBNPS SCDF was calculated as the simple average of the SCDF values for each of these frequencies, consistent with NRC General Issue-199 (GI-199) (Reference 9) and the EPRI 2025 White Paper: Seismic Penalty For RICT Calculation Approaches (Reference 10). The total DBNPS SCDF is 1.51E-5 per year. This SCDF value will be used as the conservative estimate of instantaneous SCDF ($ICDF_{seismic}$) for the DBNPS TSTF-505 LAR RICT calculations. Table E4-1 summarizes the SCDF contributions for each frequency.

3.2 Evaluate Potential Seismic Risk Increase Due to Out-of-Service Equipment

The approach taken in the computation of SCDF assumes that SCDF can be based on the likelihood that a single seismic-induced failure leads to core damage. This approach is conservative and implicitly relies on the assumption that seismic-induced failures of equipment show a high degree of correlation (i.e., if one SSC fails, all similar SSCs will also fail). This assumption is conservative, but direct use of this assumption in evaluating the risk increase from out-of-service equipment could lead to an underestimation of the change in risk. If one were to assume no correlation at all in the seismic failures, then the seismic risk would be lower than the risk predicted by a fully correlated model, but the change in risk using the un-correlated model with a redundant piece of important equipment out-of-service would be equivalent to the level predicted by the correlated model.

If the industry accepted approach (Reference 11) of correlation is assumed, the conditional core damage frequency given a seismic event will remain unaltered whether equipment is out-of-service or not. Thus, the risk increase due to out-of-service equipment cannot be greater than the total SCDF estimated by the conservative method described in Section 3.1. That is, for the DBNPS site, the delta SCDF from equipment out-of-service cannot be greater than 1.51E-5 per year.

To summarize the above considerations:

- The baseline seismic risk in this approach is assumed to be zero, whereas there will always be some level of baseline seismic risk for a zero-maintenance plant configuration. Therefore, the incremental seismic risk (configuration seismic risk vs baseline seismic risk) will always be overstated using a seismic penalty based on the total estimated seismic risk.
- The limiting HCLPF approach assumes that a failure of a component with seismic capacity at that HCLPF leads directly to core damage (CD). However, even common failure of a given set of components (e.g., all emergency diesel generators (EDGs)) would not lead directly to CD, especially in light of the post-Fukushima FLEX mitigating strategies now in place. In reality, there are few SSCs whose failure would lead to seismic CD with any significant frequency. Examples could be important structures, or the reactor pressure vessel, or “distributed systems” such as all cable trays or all piping systems.
- In a seismic PRA, seismic impacts to similar components (e.g., all the EDGs) are typically assumed to be correlated unless there are reasons to justify not correlating. Correlation has the effect of introducing common cause impacts. So, if one train of emergency AC power fails seismically, both trains are modeled as likely to fail given the same seismic event. So, in general, most seismic impacts would effectively be equivalent to TS loss of function.
- Given the above, the use of a seismic penalty based on assuming core damage given the plant level HCLPF is appropriate.

3.3 Evaluate Seismic LERF Contribution

The seismic large early release frequency (SLERF) was conservatively estimated by including the containment fragility in the convolution calculations. The seismic capability of the containment for DBNPS was evaluated in the IPEEE. The IPEEE concluded that DBNPS was seismically rugged and all components in the SPLD adequately considered the seismic input.

Since containment is one of the most rigorously seismically designed and analyzed portions of the plant, containment capability is equivalent or greater than the value assessed for plant capability. The limiting HCLPF for containment at DBNPS is 1.38g.

This HCLPF and its corresponding failure probabilities as a function of the seismic hazard level are convolved with the plant seismic hazard curves. As with the SCDF calculations, SLERF calculations are performed for frequencies of 1 Hz, 5 Hz, 10 Hz, and 100 Hz, and the simple average of these SLERF values is taken as the total SLERF for DBNPS. The total DBNPS SLERF is 1.12E-6 per year. This SLERF value will be used as the conservative estimate of instantaneous SLERF ($ILERF_{seismic}$) for the TSTF-505 submittal RICT calculations. Table E4-1 summarizes the SLERF contributions for each frequency.

Table E4-1: Seismic Penalty Results

Frequency (Hz)	SCDF (/year)	SLERF (/year)
1	1.78E-5	1.30E-6
5	1.20E-5	9.30E-7
10	1.22E-5	9.18E-7
100	1.84E-5	1.32E-6
Simple Average (1, 5, 10, 100 Hz)	**1.51E-5**	**1.12E-6**

3.4 Conclusion

The above analysis provides the technical basis for addressing the seismic-induced core damage risk for DBNPS by reducing the ICDP/ILERP criteria to account for a conservative estimate of the configuration risks due to seismic events.

The RICT and RMAT calculations are based on the discussion provided above. The actual RICT and RMAT calculations performed by the DBNPS Configuration Risk Management Tool are based on adding an incremental 1.51E-5 per year seismic CDF contribution and a corresponding 1.12E-6 per year seismic LERF contribution to the configuration-specific delta CDF and delta LERF attributed to internal and fire events contributions. This is accomplished by adding these seismic contributions to the instantaneous CDF/LERF whenever a RICT is in effect. This method ensures that an incremental seismic CDF/LERF equal to the conservative SCDF/SLERF is added to internal and fire events incremental CDF/LERF contribution for every RICT occurrence.

4. Evaluation of External Event Challenges and IPEEE Update Results

The primary purpose of this section is to address the incremental risk associated with challenges to the facility that do not exceed the design capacity. This section also provides the results of the hazard screening described earlier. Table E4-2 lists the external hazards considered.

4.1 Hazard Screening Except Seismic Events

The DBNPS IPEEE provides an assessment of the risk to DBNPS associated with external hazards. Additional analyses have been done since the IPEEE to provide updated risk assessments of various hazards, such as aircraft impacts, industrial and military facilities, and external flooding.

Table E4-2 reviews the bases for the evaluation of these hazards, identifies any challenges posed, and identifies any additional treatment of these challenges, if required. Table E4-3 provides the criteria applied in the progressive screening process used in this assessment. The conclusions of the assessment, as documented in Table E4-2, ensure that the hazard either does not present a design-basis challenge to DBNPS, or is adequately addressed in the PRA.

All hazards other than seismic activity (which is addressed via penalty analysis per Section 3), internal flooding, and internal fire (which are modeled in the DBNPS PRA) can be screened out for the DBNPS site.

In the application of RICTs, a significant consideration in the screening of external hazards is whether particular plant configurations could impact the decision on whether a particular hazard that screens under the normal plant configuration and the base risk profile would still screen given the particular configuration. The external hazards screening for DBNPS has been performed accounting for such configuration-specific impacts. The process involves multiple steps.

As a first step in this screening process, hazards that screen for one or more of the following criteria (as defined in Table E4-3) still screen regardless of the configuration, as these criteria are not dependent on the plant configuration.

- The occurrence of the event is of sufficiently low frequency that its impact on plant risk does not appreciably impact CDF or LERF. (Criterion C2)
- The event cannot occur close enough to the plant to affect it. (Criterion C3)
- The event which subsumes the external hazard is still applicable and bounds the hazard for other configurations. (Criterion C4)
- The event develops slowly, allowing adequate time to eliminate or mitigate the hazard or its impact on the plant. (Criterion C5)

The next step in the screening process would be to consider the remaining hazards (i.e., those not screened per the above criteria) to consider the impact of the hazard on the plant given particular configurations for which a RICT is allowed. For hazards for which the ability to achieve safe shutdown may be impacted by one or more such plant configurations, the impact of the hazard to particular SSCs is assessed and a basis for the screening decision applicable to configurations impacting those SSCs is provided.

As noted above, the configurations to be evaluated are those involving unavailable SSCs whose LCOs are included in the RICT Program.

Table E4-2: Evaluation of Risks From External Hazards

External Hazard[1]	Screening Result		
	Screened? (Y/N)	**Screening Criterion**	**Comment**
Aircraft Impact	Y	PS4	Bounding analysis demonstrates that the frequency of aircraft-induced radiological consequences is less than 1E-7/yr. Conservatively assuming the Large Early Release Frequency (LERF) is a surrogate for the radiological consequences and that Core Damage Frequency (CDF) is typically an order of magnitude greater than LERF for Pressurized Water Reactors, this implies that CDF is less than 1E-6/yr. Therefore, the aircraft impact hazard can be screened out from an External Events PRA for Davis-Besse Nuclear Power Station (DBNPS).
Avalanche	Y	C3	Topography is such that no avalanche is possible. The site is situated on low, flat land.
Biological Events	Y	C1 C5	The hypochlorite system inhibits growth and is controlled and monitored. There would be adequate warning for these events.
Coastal Erosion	Y	C3	DBNPS site is 3000 ft inland from the Lake Erie shoreline.
Drought	Y	C1 C4 C5	Drought could cause a reduction in the water level of Lake Erie, but the probable extreme low water level would be caused by the combination of wind tides and seiches. Drought would be a slow developing event and plant operators would have ample time to respond to drought conditions.
External Flooding	Y	C1 PS1	The updated examination of external flood risk, including the updated plant data, flood history and new measures for risk management validate the current flood mitigation strategy of the current design basis. The results of the flooding hazard reevaluation report, in conjunction with the subsequent focused flooding evaluation,

External Hazard[1]	Screening Result		
	Screened? (Y/N)	Screening Criterion	Comment
			indicate that external flooding events will cause no flooding damage to DBNPS safety-related SSCs.
Extreme Winds and Tornadoes	Y	PS1 PS4	All Seismic Category I structures are designed for the 100-year wind speed of 90 mph sustained at 30 ft above grade. Tornado loadings are based on a 300-mph rotational wind speed and a 70-mph translational wind speed, with a simultaneous maximum atmospheric pressure drop of 3 psi at a rate of 1 psi/sec. Non-Category I structures have been designed to not collapse on or impact Seismic Category I structures. DBNPS is in Region I as defined by Regulatory Guide (RG) 1.76 (Reference 12), where the tornado with an exceedance frequency of 1E-7/yr has a rotational wind speed of 184 mph, a translational wind speed of 46 mph (total wind speed of 230 mph), a pressure drop of 1.2 psi, and a pressure drop rate of 0.5 psi/sec. The DBNPS design basis tornado therefore bounds the 1E-7/yr tornado in terms of wind and pressure effects. In 2010, NRC had assessed a “Green” level Non-Cited Violation (defined as expected CDF contribution <1E-6/yr) against DBNPS following a Component Design Bases Inspection (CDBI) because part of the emergency diesel generator (EDG) exhaust stacks were not tornado missile protected, as the licensing basis had required. Also, it was discovered that the EDG Week Tank Emergency Vent was not properly missile protected. The unprotected portions of the exhaust stacks have since been removed, such that the entire piping is now missile protected. A new emergency vent system for the EDG Week Tank was installed that is missile protected. In summary, DBNPS has been designed for extreme winds and tornado loadings that are higher than the current regulatory guidance, and the plant has addressed deficiencies to comply with its licensing basis for tornado missile protection. Therefore, it is concluded that the hazard event of extreme winds and tornadoes can be screened out for DBNPS.
Internal Fire	N	None	The DBNPS Fire PRA model includes evaluation of risk from internal fires.

External Hazard[1]	Screening Result		
	Screened? (Y/N)	**Screening Criterion**	**Comment**
Fog	Y	C4 C5	It affects frequency of occurrence of other hazards, e.g., highway accidents, aircraft landing and take-off accidents and is indirectly considered. Weather forecasts can generally predict fog so there would be advance warning for this event. The mean number of days per year with heavy fog is 13 for Cleveland and 19 for Toledo.
Forest Fire	Y	C1 C4	Because the site is cleared of vegetation, and the area surrounding the site is marsh, an offsite fire would not be expected to spread directly to the site. A fire offsite could cause a loss of offsite power (LOOP), but LOOP would be addressed in the Internal Events PRA. Fires onsite would be addressed in the Fire PRA.
Frost	Y	C1 C4 C5	The effects of frost would be included in other events, such as snow and ice cover.
Hail	Y	C1 C4	Hail may occur, but there are no openings in the walls or roofs of safety-related buildings through which hail may enter and damage essential equipment. Treating hail as an external missile, it would not be the limiting external missile compared to the design basis tornado-generated missiles. The effects from hail are bounded by the current design basis of the plant.
High Tide	Y	C3	While Lake Erie is subject to wind tides, it is not subject to noticeable gravitational high or low tides, mostly due to being a much smaller body of water compared to an ocean.
High Summer Temperature	Y	C1 C5	NOAA data indicates that the all-time record high temperature for Toledo is 105°F. As discussed in the low lake level analysis, if the ultimate heat sink is lost, there is

External Hazard[1]	Screening Result		
	Screened? (Y/N)	Screening Criterion	Comment
			enough water impounded in the intake structure forebay area to provide 30 days of cooling water and maintain the plant in a safe shutdown state. UFSAR Section 9.2.5.1 documents a calculation supporting a license amendment to increase the maximum service water temperature from 85°F to 90°F. The calculation assumed a maximum ultimate heat sink temperature of 107.6°F lasting 2.3 days. The calculation demonstrated that even in a loss of coolant accident (LOCA) with no intake canal access to the lake, with the service water temperature at 90°F at initiation of the accident, there would still be a reliable 30-day supply of cooling water available. When the plant is in hot shutdown, decay heat removal normally occurs via steam bypass to the main condenser (UFSAR Section 5.1.7). Should the condenser become unavailable to perform this function, the steam can be vented to the atmosphere (UFSAR Section 5.1.10). High temperatures may contribute to a LOOP at the site, but this would be covered in the Internal Events PRA.
Hurricane	Y	C1 C4 C5	The site is well inland from any ocean and thus a landfalling hurricane would significantly weaken before reaching the site. Remnants of hurricanes can still produce intense precipitation, high winds, and even tornadoes, but these are already included in other events.
Ice Cover	Y	C1 C4 C5	Flooding at the site due to ice jams in the Toussaint River was found to be not credible.

External Hazard[1]	Screening Result		
	Screened? (Y/N)	Screening Criterion	Comment
			Ice may form on shorelines, but these layers of ice would actually be beneficial in certain ways, such as protecting the shorelines from other forms of damage, including littoral drift. Freezing rain can occur from late fall through early spring. Ice accumulation of 0.25 in or greater occurs about once a year. Ice accumulation of 0.50 in or greater occurs about once every two years. Partial freezing in the cooling tower once occurred at DBNPS during shutdown during a refueling outage. This was due to cooldown occurring too rapidly. This did not ultimately inhibit safe shutdown. Ice accumulation may cause a LOOP, but this would be covered in the Internal Events PRA.
Industrial or Military Facility Accident	Y	PS4	<u>Military Facility Accident</u> The Camp Perry Military Reservation is located 4.5 miles southeast of the site. It is used by the Ohio National Guard for small arms training, and hosts the annual National Rifle Matches. Its airspace is used for small drone training. There are no training flights or bombing runs associated with this site, so it is screened out from further consideration. <u>Industrial Facility Accident</u> The Erie Industrial Park is located approximately 4.5 miles southeast of the site. Ordnance ammunition is stored here, but the amount is not sufficient to threaten the safe operability of DBNPS in the event of an explosion. Therefore, this industrial park is screened out from further consideration.

External Hazard[1]	Screening Result		
	Screened? (Y/N)	Screening Criterion	Comment
			Flammable Stationary Sources The Erie Industrial Park stores a limited quantity of flammable material that was found by the IPEEE not to be large enough to threaten the safe operability of the site. No new data or analyses were identified that refute this conclusion. No other significant sources of offsite flammable material within 5 miles of the site were identified after the IPEEE was conducted. The IPEEE remains bounding.
Internal Flooding	N	None	The DBNPS Internal Events PRA model includes evaluation of risk from internal flooding.
Landslide	Y	C3	Terrain surrounding immediate vicinity of the site is such that a landslide is highly unlikely. The land surrounding the site is generally flat.
Lightning	Y	C1 C4	The site is designed such that safety-related equipment would be protected from lightning strikes. Lightning could cause a LOOP, but this would be covered in the Internal Events PRA.
Low Lake or River Water Level	Y	C1 C5	The lowest water level of Lake Erie at Toledo on record is 561.0 ft, International Great Lakes Datum (IGLD). The site cannot be assured of obtaining sufficient water from the lake to sustain full power operation if the lake level falls below 562 ft IGLD, and the plant would initiate shutdown if this occurs. There would be enough water impounded in the Class I excavation at the Intake Structure forebay area below 560 ft IGLD to allow the plant to safely shut down and provide at least a 30-day supply of cooling water to maintain the plant in safe shutdown condition.
Low Winter Temperature	Y	C1 C5	NOAA data indicates that the all-time record low temperature for Toledo is -20°F. It is common for Lake Erie to experience surface freezing in winter. Service water can be returned to the intake canal at the Intake Structure to prevent freezing.

External Hazard[1]	Screening Result		
	Screened? (Y/N)	Screening Criterion	Comment
			Low temperatures may contribute to a LOOP at the site, but this would be covered in the Internal Events PRA.
Meteorite/Satellite Strikes	Y	C2	This event has a very low frequency of occurrence for any site in the US.
Pipeline Accident	Y	C3	No pipelines are located within 5 miles of the site.
Intense Precipitation	Y	C4	Included under external flooding.
Release of Chemicals from On-site Storage	Y	C1	There once was a 30-ton railroad car on site that supplied chlorine for the station chlorination system, but this is no longer on site. The chlorine gas supply system was replaced with sodium hypochlorite, which is now used as the biocide. This eliminates the need for chlorine gas monitoring in the control room. No other on-site chemical storage hazards were identified that would threaten the safe operability of the plant.
River Diversion	Y	C1 C3	The mean water level of Lake Erie is not subject to variations due to diversion or source cutoffs. The site does not rely on the Toussaint River for cooling water.
Sandstorm	Y	C3	Topography is such that no sandstorm is possible.
Seiche	Y	C4	Seiches may occur in Lake Erie and may be caused by either seismic or meteorological effects. They can lead to either increased or decreased lake level at the site.

External Hazard[1]	Screening Result		
	Screened? (Y/N)	Screening Criterion	Comment
			The effects of seiches would be covered by other events such as external flooding, seismic activity, or low lake level.
Seismic Activity	N	None	While a SPRA model exists for DBNPS, the site has elected to abandon the SPRA in favor of a conservative analysis of the potential seismic impact for inclusion in the decision-making process. This analysis is described in Section 3.
Snow	Y	C1 C4	Roofs of Class I structures are designed to withstand credible snow loads. Flooding due to snow melt would be covered under external flooding.
Soil Shrink-Swell	Y	C1 C5	Investigations of surface and subsurface geotechnical conditions were conducted at the site both before and during construction. No conditions were identified that indicated soil shrink-swell would be a risk to DBNPS.
Storm Surge	Y	C4	Included in external flooding analysis.
Transportation Accidents	Y	PS4	<u>Shipments by Truck</u> The nearest highway on which explosive materials may be transported is Ohio State Route 2, which is a minimum distance of 2600 ft from any station structure. The maximum probable quantity of potentially explosive cargo for a single highway truck is 50,000 pounds (Reference 13). For this amount of explosives, the distance from the explosion where the peak positive incident overpressure will equal 1 psi is 0.31 miles, which is shorter than the distance from State Route 2 to any station structure. It is concluded that accidents along this highway with explosive cargo will not affect the safe operation of the plant. The IPEEE remains bounding. <u>Shipments by Rail</u> The nearest railroad carrying hazardous material is 5 miles south of the site. The maximum probable quantity of potentially explosive cargo for a single railroad boxcar is 132,000 pounds. For this amount of explosives, the distance from the explosion where the peak positive incident overpressure will equal 1 psi is 0.43 miles, which is

External Hazard[1]	Screening Result		
	Screened? (Y/N)	Screening Criterion	Comment
			shorter than the distance from the railroad to any station structure. It is concluded that accidents on this railroad with explosive cargo will not affect the safe operation of the plant. The IPEEE remains bounding. Shipments by Water Transport The nearest route to DBNPS by water transport is Lake Erie. The closest a ship on the lake could come to the site before the lake becomes too shallow for passage is about 8 miles. The largest probable quantity of explosive material transported by ship is approximately 5,000 tons per RG 1.91. For this amount of explosives, the distance from the explosion where the peak positive incident overpressure will equal 1 psi is 1.84 miles, which is shorter than the maximum possible distance from a ship to the site. It is concluded that shipping accidents on Lake Erie will not affect the safe operation of the plant. The IPEEE remains bounding. Potential Toxic Gas Hazards from Transient Sources The IPEEE determined that trucks carrying toxic gas hazards on State Route 2 could potentially pose a threat in the event of a major release, depending on atmospheric conditions. A model based on NUREG/CR-2650 (Reference 14) was used to determine the maximum number of trucks that can pass the site on State Route 2 carrying either explosive or toxic gaseous material that would result in a CDF contribution less than 1E-6/yr due to control room operator incapacitation, and surveys of State Route 2 were conducted to estimate the number of such trucks per day that pass the site. The surveys found that the number of trucks per day did not exceed the regulatory limit. No new regulatory guidance or data was identified that would render the IPEEE results obsolete. The IPEEE remains bounding.
Tsunami	Y	C3 C4	Included under external flooding and seismic events. Since DBNPS is on a lake, not an ocean, a tsunami large enough to impair safety-related SSCs is not expected.
Toxic Gas	Y	C4	High air pollution occurs at the site only about 1-2% of the time and does not persist for long periods when it occurs, largely due to meteorological factors such as frequent storms and winds that would disperse pollutants.

External Hazard[1]	Screening Result		
	Screened? (Y/N)	**Screening Criterion**	**Comment**
			Other toxic gas events would be included in transportation accidents, on-site chemical release, and industrial and military facility accidents.
Turbine-Generated Missiles	Y	PS4	The turbine at DBNPS has an "unfavorable" orientation as defined by RG 1.115 (Reference 15), as a turbine missile at a trajectory as low as 16 degrees above horizontal could credibly strike the Shield Building wall, if energetic enough to escape the turbine casing. The DBNPS licensing basis requires a turbine missile generation frequency of 4E-5/yr or less. DBNPS recently extended the test intervals for its turbine valves from quarterly to semi-annually. A calculation was performed to determine if the turbine missile generation frequency would remain within the licensing basis limit after the extension; the result was 3.86E-5/yr, within the limit. Combining this with updated guidance in RG 1.115 allowing for an assumed conditional probability of a turbine missile causing failure of a safety-related SSC of 1E-2 for unfavorably oriented turbines, and conservatively treating the frequency of turbine missile caused SSC failure as a CDF surrogate, the CDF due to turbine missiles is bounded at 3.86E-7/yr, less than the 1E-6/yr screening criterion.
Volcanic Activity	Y	C3	The site is not close to any active volcanoes.
Waves	Y	C4	Included in external flooding analysis.

[1] The list of hazards and their potential impacts considered those items listed in Tables D-1 and D-2 in Appendix D of RG 1.200, Rev. 3 (Reference 17)

Table E4-3: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is less than events for which plant is designed.	NUREG/CR-2300 (Reference 16) and ASME/ANS Standard RA-Sa-2009	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
	PS3. Design basis event mean frequency is <1E-5 per year and the mean conditional core damage probability is <0.1.	NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009	
	PS4. Bounding mean CDF is < 1E-6 per year.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

4.2 Seismically-Induced Loss of Offsite Power Challenges

For the DBNPS site, the only incremental risk associated with challenges to the facility that do not exceed the design capacity that is not already addressed is seismically-induced LOOP. The DBNPS seismic LOOP CDF results were conservatively estimated by including the EDG and offsite power fragility in the convolution calculations.

The internal events CDF contribution due to LOOP is 2.58E-7 per year and the estimated seismic-induced LOOP CDF contribution is 8.52E-8 per year. The ratio of the seismically induced LOOP CDF to the internal events LOOP CDF is 33%. The increase is acceptable because the seismic LOOP model used in this evaluation is conservative and does not account for additional measures to restore cooling and/or power, such as the implementation of FLEX equipment. A detailed seismic model with these options included would result in a substantially lower contribution from seismic LOOP.

Comparing the seismically induced LOOP CDF to the internal events LOOP CDF, this frequency is judged to be a sufficiently small fraction that it will not significantly impact the RICT Program calculations and can be omitted.

5. Conclusions

Based on this analysis of external hazards for DBNPS, no additional external hazards need to be added to the existing PRA models. The evaluation concluded that the hazards either do not present a design-basis challenge to DBNPS, the challenge is adequately addressed in the PRA, or the hazard has a negligible impact on the calculated RICT and can be excluded.

The ICDP/ILERP acceptance criteria of 1E-5/1E-6 will be used within the RICT Program framework to calculate the resulting RICT and RMAT based on the total configuration-specific delta CDF/LERF attributed to internal events and internal fire, plus the conservative seismic delta CDF/LERF values.

6. References

1. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,” Revision 0-A, dated October 12, 2012 (ADAMS Accession No. ML12286A322 (part of ADAMS Package Accession No. ML122860402))

2. Letter from the NRC to NEI, “Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, ‘Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines’ (TAC No.

MD4995),” dated May 17, 2007 (ADAMS Accession No. ML071200238)

3. ASME/ANS RA-Sa-2009, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” Addendum A to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009

4. NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making,” Revision 1

5. Davis-Besse Nuclear Power Station (DBNPS), Updated Final Safety Analysis Report (UFSAR), Revision 35

6. NUREG-75/087, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition,” 1975

7. NUREG-1407, “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,” June 1991

8. The Toledo Edison Company, “Individual Plant Examination of External Events (IPEEE) for the Davis-Besse Nuclear Power Station (DBNPS),” December 1996

9. NRC Generic Issue 199 (GI-199) “Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants, Safety/Risk Assessment,” dated August 2010 (ADAMS Accession No. ML100270639)

10. Electric Power Research Institute (EPRI) Report 3002032024, “Seismic Penalty for Risk Informed Completion Times (RICT) Calculation Approaches,” 2025 White Paper, September 2025

11. Letter from EPRI to the Nuclear Energy Institute (NEI), “Fleet Seismic Core Damage Frequency Estimates for Central and Eastern U.S. Nuclear Power Plants Using New Site-Specific Seismic Hazard Estimates,” dated March 11, 2014 (ADAMS Accession No. ML17268A036)

12. Regulatory Guide 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” Revision 1

13. Regulatory Guide 1.91, “Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants,” Revision 3

14. NUREG/CR-2650, “Allowable Shipment Frequencies for the Transport of Toxic Gases Near Nuclear Power Plants,” Revision 4

15. Regulatory Guide 1.115, “Protection Against Turbine Missiles,” Revision 2

16. NUREG/CR-2300, “PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants,” January 1983

17. NRC Regulatory Guide 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3, December 2020.

Enclosure 5

Davis-Besse Nuclear Power Station

License Amendment Request

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b”

Baseline CDF and LERF

Baseline CDF and LERF

1.0 Introduction

Section 4.0, Item 6 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A (Reference 2), requires that the license amendment request (LAR) provide the plant-specific total core damage frequency (CDF) and total large early release frequency (LERF) to confirm that these values are within the guidelines established in Regulatory Guide (RG) 1.174, Revision 1 (Reference 3). Note that RG 1.174, Revision 2 (Reference 4) and RG 1.174, Revision 3 (Reference 5) did not revise these guidelines.

The purpose of this enclosure is to document the technical adequacy of Davis-Besse Nuclear Power Station (DBNPS) total CDF and total LERF are within the guidelines established in RG 1.174, which does not establish firm limits for total CDF and LERF, but recommends that risk-informed applications be implemented only when the total plant risk is no more than about 1E-4/year for CDF and 1E-5/year for LERF. Demonstrating that these guidelines are met confirms that the risk metrics of NEI-06-09-A can be applied to the Risk-Informed Completion Time (RICT) program.

2.0 Baseline Risk

Table E5-1 provides the DBNPS CDF and LERF values that resulted from a quantification of the baseline internal events (which include internal flooding), fire, and seismic PRA models. Other external hazards are below accepted screening criteria and therefore do not contribute significantly to the totals. Note that the values in Table E5-1 were quantified using average maintenance (i.e., nominal maintenance was used, whereas the application specific model will be a zero-maintenance model).

Table E5-1 DBNPS Total Baseline CDF/LERF

Source	CDF Contribution (per year)	LERF Contribution (per year)
Internal Events and Internal Flooding PRA	6.25E-06	4.23E-07
Fire PRA	5.59E-05	4.93E-06
Seismic PRA	1.28E-05	9.39E-07
Total	7.50E-05	6.29E-06

Table E5-2 provides similar results as Table E5-1 with the exception that the seismic penalty is applied instead of the seismic PRA.

Table E5-2 DBNPS Total Baseline CDF/LERF (with Seismic Penalty)

Source	CDF Contribution (per year)	LERF Contribution (per year)
Internal Events and Internal Flooding PRA	6.25E-06	4.23E-07
Fire PRA	5.59E-05	4.93E-06
Seismic Penalty	1.51E-05	1.12E-06
Total	7.73E-05	6.47E-06

As demonstrated in both Table E5-1 and Table E5-2, the total CDF and total LERF are within the guidelines set forth in RG 1.174 and support small changes in risk that may occur during RICT entries following TSTF-505 implementation. Therefore, DBNPS TSTF-505 implementation is consistent with NEI 06-09-A guidance.

3.0 References

1. Letter from the NRC to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)", dated May 17, 2007 (ADAMS Accession No. ML071200238)
2. NEI Topical Report NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines", Revision 0, dated October 2012 (ADAMS Accession No. ML12286A322)
3. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", Revision 1, November 2002, (ADAMS Accession No. ML023240437)
4. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", Revision 2, May 2011, (ADAMS Accession No. ML100910006)
5. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", January 2018, Revision 3, (ADAMS Accession No. ML17317A256)

Enclosure 6

Davis-Besse Nuclear Power Station

License Amendment Request

Revise Technical Specifications to Adopt RISK Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b"

Justification of Application of At-Power PRA Models to Shutdown Modes

Justification of Application of At-Power PRA Models to Shutdown Modes

This enclosure is not applicable to the Davis-Besse Nuclear Power Station submittal. Vistra is proposing to apply the Risk-Informed Completion Time Program only in Modes 1 and 2 and not in the shutdown Modes.

Enclosure 7

Davis-Besse Nuclear Power Station

License Amendment Request

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b”

PRA Model Update Process

PRA Model Update Process

1.0 Introduction

Section 4.0, Item 8 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A (Reference 2) requires that the license amendment request (LAR) provide a discussion of the licensee's programs and procedures which assure the PRA models that support the Risk-Managed Technical Specifications (RMTS) are maintained consistent with the as-built, as-operated plant.

Vistra utilizes a formal process to maintain and update the PRA models such that the models reflect the as-built, as-operated plant. The PRA models of record will be used as the basis for the application specific model that supports the Risk-Informed Completion Time (RICT) program at the Davis-Besse Nuclear Power Station (DBNPS). This enclosure describes the procedural processes that will be used to ensure the configuration control of models used to support the RICT program. Plant changes, including physical modifications and procedure revisions, will be identified and reviewed prior to implementation to determine if the change could impact the PRA models. Vistra's PRA configuration control process is governed by procedure FLT-CC-PM6000, Probabilistic Risk Assessment Program (Reference 3). The process ensures that plant modifications, procedure revisions, and other items are incorporated into the PRA models as appropriate. The process will include monitoring for and evaluation of conditions that may impact the PRA models (including potential or known errors); issues that are determined to be conditions adverse to quality will be entered into the station's Corrective Action Program.

If it is identified that a plant change or discovered condition has a significant impact to RICT program calculations as defined by procedure, then a PRA model interim revision will be implemented. Otherwise, the change will be incorporated during the next periodic model update. Such pending changes are considered when evaluating other changes until they are fully implemented into the PRA models. Periodic updates will typically be performed every two fuel cycles.

2.0 PRA Model Update Process

2.1 <u>Internal Events, Internal Flooding, Fire, and Seismic Model Maintenance and Update</u>

The Vistra fleet PRA model maintenance and update process will ensure that the applicable PRA models of record and application specific model used for the RICT program reflect the as-built, as-operated plant at DBNPS. The process delineates the responsibilities and guidelines for updating the full power internal events, internal flooding, fire, and seismic PRA models, and includes both periodic and interim updates.

The process will include provisions to:

- track, evaluate, and prioritize issues that may affect the technical elements of the PRA models (e.g., due to plant changes, plant/industry operational experience, or errors or limitations identified in the model);
- assess the individual and cumulative risk impact of unincorporated changes; and

- control the model and necessary computer files, including those associated with the Effective Reference Model (ERM).

Industry best practices and consensus modeling techniques are also reviewed and monitored to ensure Vistra's PRA is using state-of-the-art processes and methods. Changes that are considered an upgrade per the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by Regulatory Guide 1.200, Revision 3) will receive a peer review focused on those aspects of the PRA model that represent the upgrade.

2.2 Review of Plant Changes for Incorporation into the PRA Model

1. Plant changes or discovered conditions are reviewed for potential impact to the PRA models, including the real rime risk (RTR) model and the subsequent risk calculations which support the RICT Program (NEI 06-09-A, Section 2.3.4, Items 7.2 and 7.3, and 2.3.5, Items 9.2 and 9.3).
2. Plant changes that meet the criteria defined in References 3 and 4 (including consideration of the cumulative impact of other pending changes) will be incorporated in the applicable PRA model(s), consistent with the NEI 06-09-A guidance. Otherwise, the change is assigned a priority and is incorporated at a subsequent periodic update consistent with procedural requirements (NEI 06-09-A, Section 2.3.5, Item 9.2).
3. PRA updates for plant changes are performed at least once every two refueling cycles, consistent with the guidance of NEI 06-09-A (NEI 06-09-A, Section 2.3.4, Item 7.1, and 2.3.5, Item 9.1).
4. If a PRA model change is required for the RTR model, but cannot be immediately implemented for a significant plant change or discovered condition, either:
    a. Interim analysis to address the expected risk impact of the change will be performed. In such a case, these interim analyses become part of the RICT Program calculation process until the plant changes are incorporated into the PRA model during the next update.
    OR
    b. Appropriate administrative restrictions on the use of the RICT Program for extended Completion Times are put in place until the model changes are completed, consistent with the guidance of NEI 06-09-A.

These actions satisfy NEI 06-09-A, Section 2.3.5, Item 9.3.

3.0 References

1. Letter from the NRC to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)", dated May 17, 2007 (ADAMS Accession No. ML071200238)
2. NEI Topical Report NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines", Revision 0, dated October 2012 (ADAMS Accession No. ML12286A322)
3. Probabilistic Risk Assessment Program, FLT-CC-PM6000, Revision 0
4. Probabilistic Risk Assessment Model Management, FLT-CC-BP6001, Revision 0

Enclosure 8

Davis-Besse Nuclear Power Station

License Amendment Request

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b”

Attributes of the Real-Time Model

Attributes of the Real-Time Model

1.0 Introduction

Section 4.0, Item 9 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A (Reference 2), requires that the license amendment request (LAR) provide a description of the PRA models and tools used to support the Risk-Managed Technical Specifications (RMTS). This includes identification of how the baseline PRA models will be modified for use in the Configuration Risk Management Program (CRMP) tools, quality requirements applied to the PRA models and CRMP tools, consistency of calculated results from the PRA models and CRMP tools, and training and qualification programs applicable to personnel responsible for development and use of the CRMP tools. This item also requires confirmation that the CRMP tools can be readily applied for each Technical Specification (TS) Limiting Condition for Operation (LCO) within the scope of the plant-specific submittal.

This enclosure describes the necessary changes to the peer-reviewed baseline PRA models for the use in the CRMP software, which supports the Risk-Informed Completion Time (RICT) program. The process that will be employed to adapt the baseline models is demonstrated:

1. To preserve the core damage frequency (CDF) and large early release frequency (LERF) quantitative results;
2. To maintain the quality of the peer-reviewed PRA models; and
3. To correctly accommodate changes in risk due to configuration-specific considerations.

Controls for ensuring PRA technical adequacy and training programs applicable to the RICT program are also discussed in this enclosure.

2.0 Translation of Baseline Model for the Use in Configuration Risk

The baseline PRA models for internal events, internal flooding, and internal fire are peer-reviewed models. These models are updated when necessary to incorporate changes to reflect the as-built, as-operated plant as discussed in Enclosure 7. The application specific model used for the RICT program will be a zero-maintenance model capable of calculating internal events, internal flooding, and fire risk for each plant configuration. These models will be merged into a single “one-top” model for use in the CRMP. The CRMP model, which is used to implement the RICT program, will be verified to provide results equivalent to the baseline models in accordance with approved procedures.

The CRMP software will be used to facilitate all configuration-specific risk calculations and support RICT program implementation. The CRMP software is designed to quantify the risk from internal events, internal flooding, and fire and then use those values to calculate the risk management action time (RMAT) and RICT for each configuration. The RMAT and RICT will also account for the risk from seismic activity by applying a conservative CDF and LERF penalty to each calculation.

The treatment of common cause failure (CCF) will be in accordance with the approach described in NEI 06-09. For planned RICTs (e.g., to perform preventive maintenance tasks), no changes in CCF factors would be made in the CRMP model since no failures have occurred. For emergent failures, Operations personnel would perform an extent of condition evaluation (using existing plant processes) to determine if any CCF potential exists. If it is determined that a CCF model impacts redundant equipment and results in a TS loss of function, the RICT program would not apply. If a potential CCF is determined to not result in a loss of function, then a quantitative or qualitative evaluation of the impact on the RICT would be performed. If a quantitative evaluation of increased CCF probability is performed, the adjustment to CCF probability will be made in accordance with Regulatory Guide 1.177, "Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications", Revision 2 (Reference 3). If a qualitative evaluation is performed, Risk Management Actions (RMAs) to manage a possible CCF would be considered for implementation. See Enclosure 12 for additional information regarding the development of RMAs with consideration of CCF modes.

3.0 Quality Requirements and Consistency of PRA Model and Configuration Risk Management Tools

The approach for establishing and maintaining the technical adequacy of the PRA models, including the CRMP model, includes both a PRA maintenance and update process (described in Enclosure 7), and the use of self-assessments and independent peer reviews (described in Enclosure 2).

The information provided in Enclosure 2 demonstrates that the Davis-Besse Nuclear Power Station (DBNPS) internal events, internal flooding, and fire PRA models conform to the associated industry standards endorsed by Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Revision 3 (Reference 4). This information provides a robust basis for concluding that the PRA models are technically adequate for use in risk-informed licensing actions.

For maintenance of an existing CRPM model, changes made to the baseline PRA model in translation to the CRMP model will be controlled and documented in accordance with Vistra's PRA configuration control procedure FLT-CC-BP6001 (Reference 5). The process will ensure models are accurate, as described in Enclosure 7. Because the CRMP model is developed from the complete baseline PRA models (i.e., it is not a simplified model), the results of this model would be expected to be essentially identical to those of the constituent baseline PRA models for internal events, internal flooding, and fire hazards (after accounting for the items such as setting maintenance events to zero). Acceptance testing will be performed after every CRMP model update to ensure that the software functions as intended and that quantification results are reasonable.

These actions satisfy NEI 06-09-A, Section 2.3.5, Item 9.

4.0 Training and Qualifications

The PRA staff is responsible for development and maintenance of the CRMP model. Operations and Work Control staff will use the configuration risk tool to implement the RICT program. The PRA and Operations staff are trained in accordance with a program using National Academy for Nuclear Training ACAD documents, which is also accredited by the Institute of Nuclear Power Operations (INPO).

5.0 Application of Configuration Risk Tool to the RICT Program Scope

The Electric Power Research Institute (EPRI) PHOENIX software, or equivalent, will be used to facilitate configuration-specific risk calculations and support the RICT program implementation. This software is specifically designed to support the implementation of RMTS. The PHOENIX software will permit the user to evaluate all plant configurations using appropriate mapping of plant equipment to the PRA basic events. The equipment in the scope of the RICT program will be able to be evaluated in the appropriate PRA models. See Enclosure 1 for additional information regarding PRA functions and corresponding LCO Required Actions.

6.0 References

1. Letter from the NRC to NEI, “Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines’ (TAC No. MD4995),” dated May 17, 2007 (ADAMS Accession No. ML071200238)
2. NEI Topical Report NEI 06-09-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,” Revision 0, dated October 2012 (ADAMS Accession No. ML12286A322)
3. NRC Regulatory Guide 1.177, “Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,” Revision 2, dated January 2021 (ADAMS Accession No. ML20164A034)
4. NRC Regulatory Guide 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 3, dated December 2020 (ADAMS Accession No. ML20238B871)
5. FLT-CC-BP6001, “Probabilistic Risk Assessment Model Management,” Revision 0

Enclosure 9

DAVIS BESSE NUCLEAR POWER STATION

License Amendment Request

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b"

Key Assumptions and Sources of Uncertainty

Key Assumptions and Sources of Uncertainty

1.0 Introduction

The purpose of this enclosure is to disposition the impact of Probabilistic Risk Assessment (PRA) modeling epistemic uncertainty for the Risk Informed Completion Time (RICT) Program. The NRC final Safety Evaluation for NEI Topical Report NEI 06-09-A (Reference 1), Section 4.0, Item 10 requires a discussion of how the key assumptions and sources of uncertainty were identified, and how their impact on the RMTS was assessed and dispositioned, such as through Risk Management Actions (RMA). The baseline internal events PRA (including Level 2), internal flood and fire PRA models document assumptions and sources of uncertainty and these were reviewed during the model peer reviews. Therefore, the approach taken is to review these documents to identify the items that may be directly relevant to the RICT Program calculations, to perform sensitivity analyses where appropriate, to discuss the results, and to provide dispositions for the RICT Program. The seismic penalty approach was used for the Davis Besse RICT application. The method produces conservative seismic penalty results for seismic CDF and seismic LERF that result in more restrictive RICT times. Therefore, no further evaluation of the impact of seismic penalty conservatism is included in this enclosure.

The epistemic uncertainty analysis approach described in section 2.0 applies to the internal events (including Level 2), internal flooding, and fire PRAs. In addition, NEI 06-09-A requires that the uncertainty be addressed in RICT Program Real Time Risk tools by consideration of the translation from the PRA model. The Real Time Risk model, also referred to as the Configuration Risk Management Program (CRM) model, discussed in Enclosure 8 of this license amendment request (LAR), will include internal events, internal flooding events, and fire events. The model translation uncertainties evaluation and impact assessment are limited to new uncertainties that could be introduced by application of the CRM tool during RICT Program calculations.

Assumptions and sources of uncertainty for all utilized PRA models were reviewed to identify those which could be significant for the evaluation of this application. If the DBNPS PRA Model used a non-conservative treatment or methods that are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine its impact on this application. To identify these assumptions and sources of uncertainty, both plant specific and generic sources of uncertainty were considered. All PRA notebooks were reviewed, and sources of uncertainty were compiled and characterized in the applicable DBNPS notebooks. The identification and characterization of the sources of uncertainty was performed consistent with the requirements of the ASME/ANS PRA Standard, ASME/ANS RA-Sa-2009 (Reference 3). This evaluation meets the intent of steps C-1 and E-1 of NUREG-1855, Revision 1 (Reference 4). To assess the impact of sources of uncertainties on the TSTF-505 application, a review of the base model sources of uncertainty for the DBNPS PRA models was performed. Each identified uncertainty was evaluated with respect to its potential to significantly impact the decisions of this submittal. This evaluation meets the intent of the screening portion for steps C-2 of and E-2 of NUREG -1855.

The process used to evaluate sources of uncertainty for the RICT application follows the guidance illustrated in Figure 4-1 of EPRI TR-1016737. Tables E9-1 through E9-4 in Section 2.0 of this enclosure lists plant specific and generic sources of uncertainty that were found to potentially impact risk informed applications in general. Those potentially impacting the RICT application are noted in the final column of the tables as recommended for sensitivities on the RICT quantification.

2.0 Assessment of Internal Events, Flooding and Fire PRA Epistemic Uncertainty Impacts

To identify key sources of uncertainty, the internal events (Including Level 2), flooding and fire PRA models uncertainty documentation was developed, based on the guidance in NUREG-1855 (Reference 4) and Electric Power Research Institute (EPRI) 1016737 (Reference 2). As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail uncertainties).

Parametric uncertainty was addressed as part of the DBNPS baseline PRA model quantification and is documented in the Internal events Quantification notebook (Reference 4) and the Fire sensitivity and uncertainty notebook.

Model uncertainty arises because different approaches exist to represent plant response. A *source of model uncertainty* is one related to an issue in which no consensus approach or model exists, and where the choice of approach or model is known to have an effect on the PRA. These uncertainties are typically dealt with by making assumptions (e.g., the approach to address common-cause failure, how a RCP would fail following a loss of seal cooling, the approach to identify and quantify HFEs). The guidance provided in EPRI 1016737, *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments* (Reference 2), was used to address sources of model uncertainty and related assumptions. It provides a framework for the pragmatic treatment of uncertainty characterization to support risk informed applications and decision making. The process includes identification and characterization of sources of model uncertainty and related assumptions. The Electric Power Research Institute (EPRI) compiled a listing of generic sources of modeling uncertainty to be considered for each PRA technical element (Reference 2) and the evaluation performed for DBNPS considered each of the generic sources of modeling uncertainty as well as the plant-specific sources. Additionally for the Level 2 PRA, DBNPS addresses each of the LERF related sources of uncertainty listed in EPRI TR-1009652 (Reference 24).

Completeness uncertainty relates to risk contributors that are not in the PRA model. These include known types such as the scope of the PRA, which does not include some classes of initiating events, hazards, and operating modes; and the level of analysis, which may have omitted phenomena, failure mechanisms, or other factors because their relative contribution is believed to be negligible. They also include ones that are not known such as the effects on risk from aging or organizational changes; and omitted phenomena and failure mechanism that are unknown. Both can have a significant impact on risk. No completeness uncertainties were identified for the Davis-Besse Level 1 PRA, based on the ASME/ANS PRA Standard (Reference 3). Regarding the fire PRA, the fire sensitivity and uncertainty report (Reference 8) identifies and characterizes the uncertainties associated with the Davis Besse Fire PRA process in accordance with NUREG/CR-6850 (Reference 10) Task 15 which includes consideration of completeness uncertainties.

A detailed evaluation of assumptions and sources of uncertainty for the IE PRA model, Level 2, Internal flood, and fire PRA models are included in References 5, 6, 7 ,and 8. A review of these assumptions and sources of uncertainty was performed to identify those that might impact PRA applications in general. The identified items are documented and evaluated to determine if they may be a key source of uncertainty for the RICTs. The DBNPS Impact of Model Uncertainty on the RICT Process report (Reference 9) documents this review.

The calculation of a RICT is based on Incremental Core Damage Probability (ICDP) and Incremental Large Early Release Probability (ILERP). These are delta-risk measures that evaluate the change in risk over the baseline “zero maintenance” risk for the plant. In reviewing each of the candidate sources of uncertainty for the internal events, internal flooding, and fire PRAs, the following considerations were applied to determine if a RICT impact could exist. If a candidate source of uncertainty could be shown to satisfy these considerations, then this was considered to be adequate.

- Criterion #1: Candidate uncertainties that are qualitatively shown to have a very small impact on total risk and would be expected to have a negligible impact on delta-CDF and delta-LERF (particularly uncertainties that pertain to parts of the model that would not impact components that are in the RICT program, such as changes to non-support system initiating event frequencies, human error probabilities not related to RICT-eligible equipment, etc.).
- Criterion #2: Candidate uncertainties that are represented through conservative PRA modeling that would be expected to have a negligible or conservative impact on delta-risk RICT calculations.
- Criterion #3: Candidate uncertainties that were identified, but for which current industry-accepted approaches and data were used, are not considered as key sources of uncertainty. This is consistent with the ASME/ANS PRA Standard definition of a “source of modeling uncertainty” which states: “a source is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model.”
- Criterion #4: Candidate uncertainties that were examined via sensitivity studies to confirm that the impact on baseline CDF and LERF are negligibly small are not considered as key sources of uncertainty for the RICT program.

A total of 25 generic and plant specific uncertainties were identified as potentially impacting PRA applications. After review of these items and applying the screening criteria above, 2 of these uncertainties remained as potentially impacting the RICT process (Reference 9). These are listed below:

Source of Uncertainty Topic	Description	Additional Information
Critical Flood Height	Critical flood height related assumptions could result in conservative flood mitigation HEPs and cause an overestimation of the extent of target damage, resulting in an underestimation of delta risk for mitigating components.	Table E9-2, Item # 2
Credit for non-standard success paths (FLEX)	Uncertainty associated with crediting non-standard success paths.	Table E9-4, Item # 6

A review of these uncertainty topics was performed and documented in Appendix F of the Evaluation of RICT Durations for Davis-Besse Nuclear Power Station (Reference 11). The sensitivities performed for these items concluded that none of the assumptions or uncertainties could impact the RICT calculations in a non-conservative manner.

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
1	RCP seal LOCA treatment–PWRs (Item #5 of Table M-1 of Reference 5)	The assumed timing and magnitude of RCP seal LOCAs given a loss of seal cooling can have a substantial influence on the risk profile. Failure and timing for the DBNPS seal LOCA model are based on the Byron Jackson N-9000 seal design. Failure modeled due to loss of return or all cooling, and failing to trip pumps. This affects Accident sequences involving loss of seal cooling. Transient-induced seal LOCAs are important contributors to risk.	Because Byron Jackson (BJ) N-9000 seals are used at Davis Besse, the failure and timing for its seal LOCA model are based on the BJ N-9000 design, the identified assumptions represent realistic modeling. Therefore these do not require further evaluation.	Per industry practice, the timing and magnitude of RCP seal LOCAs for Davis Besse are based on plant specific design information. Therefore, per Criterion #3, this is not a source of uncertainty for the RICT application. Also, it should be noted that there is less uncertainty in the resulting end-states associated with the BJ seal model, than the uncertainty associated with other seal models that use film riding seals.	This item does not represent a key source of uncertainty for the RICT process.

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
2	Room heat up calculations (Item #6 of Table M-1 of Reference 5)	Loss of HVAC can result in room temperatures exceeding equipment qualification limits. Treatment of HVAC requirements varies across the industry and often varies within a PRA.	In general, modeling of HVAC could have a realistic and/or conservative bias; may be a slight source of model uncertainty in some applications.	Guidance in NEI 06-09-A (Reference 1) states that if the PRA model is constructed using data points or basic events that change as a result of time of year or time of cycle, then the RICT calculation shall either use the more conservative assumption at all times or be adjusted appropriately to reflect the current (e.g., seasonal or time of cycle) configuration for the feature as modeled in the PRA. Otherwise, time averaged data may be used in establishing the RICT." Based on the split fractions discussion in the Davis Besse HVAC PRA notebook (Reference 23) the model includes house events associated with air temperatures, allowing for the assignment of frequencies to various temperature ranges. This satisfies the "time-averaged data" approach allowed for in NEI-06-09-A. Therefore, the impact of seasonal variation on HVAC is addressed in the model. Per Criterion #1, this is not a key source of uncertainty.	This item does not represent a key source of uncertainty for the RICT process.

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
3	Unavailability Assigned to Standby Train (Item #24 of Table M-2 of Reference 5)	For systems that include a normally running train and a standby train, data was collected for both trains, summed, and all the unavailability was applied to the standby train.	This modeling approach could result in overstating the vulnerability of the standby train and understating the risk associated with the primary train. This could lead to misleading conclusions about system reliability and risk contributors.	Per the (RICT) program in accordance with the guidance in NEI 06-09 (Reference 1), maintenance unavailability events are zeroed out for the RICT calculations, so the assignment of unavailability to a particular train will have no impact on the RICT application. Although an official sensitivity analysis was not performed, this uncertainty is not a key source of uncertainty, as per Criterion #4.	This item does not represent a key source of uncertainty for the RICT process.
4	SGTR HRA Timing (Item #21 of Table M-2 of Reference 5)	RCS carryover into the secondary side is assumed to occur during an SGTR when the safeties lift. This conservative and bounding estimate is used in the HRA.	This modeling approach is a conservatism in the analysis and could potentially impact risk application results.	If the effected SGTR HFEs are present in the RICT delta cut sets, then the conservative HEPs would result in more restrictive completion times. If the SGTR HFEs are not present in the RICT delta cut sets, there would be no impact. Therefore this is not a key source of uncertainty, as per Criterion #2.	This item does not represent a key source of uncertainty for the RICT process.

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
5	MGL CCF Parameter Estimates (Item #25 of Table M-2 of Reference 5)	NRC and WCAP data used for CC parameters. The alpha and MGL distributions were assumed to be beta distributions; the beta distribution variance is conserved, so the MGL variance is assumed to equal the alpha variance; the MGL a and b values are calculated based on the MGL mean and alpha variance; if the sum value of the MGL non-adjusted a and b is greater than 10,000, the alpha range scales down the □ and □ factors factor to produce the adjusted A (a Adj) and adjusted B (b Adj) factors.	CCF is an understood and accepted uncertainty for applications.	Potentially, all LCOs in the RICT program could be affected by uncertainties related to CCF. An industry-accepted approach is used in the modeling for common cause failure events and is not considered a key uncertainty for the RICT program per Criterion #3. Furthermore, it is expected that in the RICT quantification for planned maintenance, CCFs will be present in both the base configuration and the RICT configuration, which will largely cancel each other out in the delta risk calculation. This is not necessarily the case for emergent RICTs, where the failure is associated with a common cause issue that impacts other trains. In such cases, no RICT would be allowed.	This item does not represent a key source of uncertainty for the RICT calculations.

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
6	CCF for Large Groups (Item #26 of Table M-2 of Reference 5)	A simplification was used for large CC groups (i.e., more than four components). The combinations of two were modeled; however, the beta factor used was based on the total number of components in the group (or largest available).	CCF is an understood and accepted uncertainty for applications.	Potentially all LCOs in the RICT program could be affected by uncertainties related to CCF. An industry-accepted approach is used in the modeling for common cause failure events and is not considered a key uncertainty for the RICT program per Criterion #3. Furthermore, it is expected that in the RICT quantification for planned maintenance, CCFs will be present in both the base configuration and the RICT configuration, which will largely cancel each other out in the delta risk calculation. This is not necessarily the case for emergent RICTs, where the failure is associated with a common cause issue that impacts other trains. However, in such cases, no RICT would be allowed.	This item does not represent a key source of uncertainty for the RICT calculations.
7	HFE Optimum and High Stress (Item #27 of Table M-2 of Reference 5)	For all pre-initiating events, *optimum stress* is applied due to the level of experience, the nature of the event, and lack of being unduly challenged in performing the proceduralized tasks in a normal operating environment. Most post-initiator HRA events use *optimum stress* due to the level of experience, the nature of the event, and lack of being unduly challenged in performing the proceduralized tasks; some use *high stress* (e.g., realigning the SBODG to the MDFP).	HRA is an understood and accepted uncertainty for applications.	Potentially, all LCOs in the RICT program could be affected by uncertainties related to HFEs. The process used was consistent with Industry Practice. Per Criterion #3, this is not a key uncertainty for the RICT application. Also, the presence of affected HEPs in both the base and RICT configurations results in their impacts essentially canceling each other out in the delta risk calculation, thereby supporting the exclusion of this uncertainty from further evaluation.	This item does not represent a key source of uncertainty for the RICT calculations

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
8	HFE Staffing (Item #28 of Table M-2 of Reference 5)	HFE staffing considers a Control Room Supervisor, RO, and Balance of Plant Operators who are normally in the Control Room, a Shift Supervisor, and a Shift Engineer (who is also the STA) on each operating crew. For non-time-critical actions, a *recovery factor* for the extra crew member is generally credited. Credit for STA actions are generally not assumed until 15 minutes after the initiating event occurs.	HRA is an understood and accepted uncertainty for applications.	Potentially all LCOs in the RICT program could be affected by uncertainties related to HFEs. The process used was consistent with Industry Practice. Per Criterion #3, this is not a key uncertainty for the RICT application.	This item does not represent a key source of uncertainty for the RICT calculations
9	HFE Timing (Item #29 of Table M-2 of Reference 5)	Timing for various scenarios is based on simulator observations and/or MAAP analysis.	HRA is an understood and accepted uncertainty for applications.	Potentially all LCOs in the RICT program could be affected by uncertainties related to HFEs. The process used was consistent with Industry Practice. Per Criterion #3, this is not a key uncertainty for the RICT application. Also, the presence of affected HEPs in both the base and RICT configurations results in their impact largely canceling out in the delta risk calculation, thereby supporting the exclusion of this uncertainty from further evaluation.	This item does not represent a key source of uncertainty for the RICT calculations

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
10	HFE Dependency and Modeling (Item #30 of Table M-2 of Reference 5)	No lower limit (i.e. floor) was used for combinations of post-initiator HFEs (Reference 5).	No impact; More recent documentation provided in the Human Reliability Analysis Notebook (Reference13) states that a floor of 1E-05 was applied with exceptions, based in industry guidance in EPRI 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events." (Reference 14)	N/A; no impact on the applications.	This item does not represent a key source of uncertainty for the RICT application.
11	Identification of HFEs (Item #31 of Table M-2 of Reference 5)	Systematic processes were applied to ensure that human interactions that could be important to risk were identified. These processes are judged sufficient to ensure that the most important ways in which humans may have an influence on an accident are identified.	HRA is an understood and accepted uncertainty for applications.	Potentially, all LCOs in the RICT program could be affected by uncertainties related to HFEs. Systematic processes were applied to ensure that human interactions that could be important to risk were identified. The process used was consistent with Industry Practice. Therefore, this uncertainty meets Criterion #3 and no additional sensitivity for the RICT process is needed.	This item does not represent a key source of uncertainty for the RICT calculations

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
12	HFE Characterization and Probabilities (Item #32 of Table M-2 of Reference 5)	The probabilities for HFEs are estimated based on models that have reached a point at which there is general agreement among the industry. There are uncertainties in the inputs such as the time available for the operators to act under various conditions, which are judged to be small. Table M-2, Item 32)	HRA is an understood and accepted uncertainty for applications.	Potentially, all LCOs in the RICT program could be affected by uncertainties related to HFEs. The process used was consistent with Industry Practice. Per Criterion #3, this is not a key uncertainty for the RICT application. Also, the presence of affected HEPs in both the base and RICT configurations results in their impacts largely canceling each other out in the delta risk calculation, thereby supporting the exclusion of this uncertainty from further evaluation.	This item does not represent a key source of uncertainty for the RICT calculations
13	Support System Initiating Events (Item #2 of Table M-1 of Reference 5)	Increasing use of plant-specific models for support system initiators (e.g., loss of SW, CCW, or IA, and loss of ac or dc buses) have led to inconsistencies in approaches across the industry. A number of challenges exist in modeling of support system initiating events: 2a. Treatment of CCFs 2b. Potential for recovery It was assumed that the use of generic CCF factors is applicable. Fault tree models were used for several support system IEs. CCFs were based on WCAP and NRC generic CCF data, and EPRI 1013490 guidance for CCF factor adjustments. Potential for recovery is credited in some of the support	CCF is an understood and accepted uncertainty for applications. Conservative recovery action values credited in support system initiating event fault trees could result in overestimation of some initiating event frequencies.	Potentially, all LCOs in the RICT program could be affected by uncertainties related to CCF. An industry-accepted approach is used in the modeling for common cause failure events and is not considered a key uncertainty for the RICT program per Criterion #3. Furthermore, it is expected that, in the RICT quantification for planned maintenance, CCFs would be in both the base and RICT configurations, largely canceling each other out in the delta risk calculation. This is not necessarily the case for emergent RICTs, where the failure is associated with a common cause issue that impacts other trains. In such cases, no RICT would be allowed. Use of screening values for recovery actions is consistent with industry practice and therefore is	This item does not represent a key source of uncertainty for the RICT calculations.

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
		system initiating event frequencies. These recovery actions are conservative screening values.		not considered a key source of uncertainty per Criterion #3. Also, it is expected for RICT quantification that the initiating events, potentially overestimated due to conservative recovery action values, would be in both the base and RICT configurations and would cancel each other out in the delta risk calculation. Therefore, this is not considered a key uncertainty for the RICT program per Criterion #2.	
14	LOCA Initiating Event Frequencies (Item #3 of Table M-1 of Reference 5)	It is difficult to establish values for events that have never occurred or have rarely occurred with a high level of confidence. The choice of available data sets or the use of specific methodologies in the determination of LOCA frequencies could impact the base model results and some applications.	The DBNPS PRA used data from NUREG/CR-6928 to calculate LOCA initiating event frequencies, including pipe break frequencies.	Because this approach is consistent with industry practice, it does not require a sensitivity evaluation based on Criterion #3. Also, this uncertainty topic pertains to a non-support system initiating event frequency and therefore screens based on Criterion #1.	This item does not represent a key source of uncertainty for the RICT calculations.

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
15	In-core melt Progression (Uncertainty designator LE-B-5, Table 7.5-2 of Reference6)	MAAP is used to characterize the phenomena of core melt progression and fission product transport. Assumptions associated with 1) the induced SGTR (ISGTR), specifically probability of a stuck open SG relief valve based on 30 demands in ISGTR sequences 2) how the progression affects timing of hot leg/surge line rupture 3) the probability of failure of in-vessel recovery of molten core with successful low pressure injection, recirculation and containment heat removal were identified as potential sources of uncertainty.	Sensitivities addressing items 1) and 3) in the "Summary of Assumption/Uncertainty" column were documented in sections 8.5 and 8.7 of Reference 6. The sensitivity for item 2), addressing the timing of hot leg/surge line rupture, varied the probability of induced hot leg failure to understand the impact on LERF versus Large Late release frequency. The sensitivity (Section 8.6 of Reference 6 acknowledges conservatism in the selected failure probability but does not specify the impact on the overall LERF results. Based on inspection, the impact is judged to be less than 1% impact on LERF.	Based on a review of the associated sensitivities, this uncertainty meets Criterion #1 and Criterion #4, and no additional sensitivity analysis for the RICT process is required.	This item does not represent a key source of uncertainty for the RICT calculations.

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
16	Ex-vessel core melt progression – Debris impingement (Uncertainty designator LE-B-7, Table 7.5-2 of Reference 6)	Regarding debris impingement on the containment shell, there is some uncertainty and MAAP calculations are not sufficient. There is substantial uncertainty both in the degree of debris entrainment at various RCS pressures and in the likelihood that an uncoolable debris pool would form at the area outside the cavity exit that could eventually lead to a liner attack. A sensitivity of the sidewall failure assumptions is performed in Section 8.10.	The sensitivity analysis documented in Section 8.10 of Reference 6 examines the impact of varying the sidewall failure probability and based on the impact on the results, the probability in the model was found to be appropriate. It should be noted that varying the sidewall failure probability from 1E-2 to 1E-1 resulted in a LERF increase of 23.3% Reference 6 concludes that this is a minor risk increase given that the failure probability was increased by 900%.	The applicable sensitivity study concluded that the value assumed for the sidewall failure probability acceptable. Therefore, this uncertainty candidate can be screened based on Criterion #4.	This item does not represent a key source of uncertainty for the RICT calculations.

Table E9-1: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Events, including Level 2)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
17	Steam Generator Atmospheric Vent Valve (AVV) (Sections 5.2.3 and 5.2.6.4 of Table 7.5-3 of Reference 6)	The conservative assumption is made that all SGTR sequences with an AVV that remains open during the core melt progression (as opposed to opening in the Level 1 event tree) will be considered a LERF. The probability of a Steam Generator AVV sticking open during the accident progression was assessed assuming that there would be 30 demands to open. In addition, the conditional probability of a relief valve failing to reseat, given successful reseat after the first demand, was modeled as 3.93E-4.	The assumption that all SGTR sequences with an AVV that sticks open during the core melt progression will be considered a LERF applies to Plant Damage States 67 and 68, which, combined, contribute to 1.66E-10/yr to the overall LERF. This has a less than 1% impact on LERF and is therefore considered to not affect risk applications. The sensitivity (Section 8.5 of Reference 6) addressing the assumption of 30 demands for the probability of an SG AVV sticking open and the conditional probability of a relief valve failing to reseat resulted in a <0.005% change in LERF and therefore, is considered not to have an impact on risk applications.	Based on the review of the associated sensitivities, these uncertainties are screened based on Criterion #1, and no additional sensitivity analysis for the RICT process is needed.	This item does not represent a key source of uncertainty for the RICT calculations.

Table E9-2: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Flood PRA)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
1	Flood Water Spill Rate (Section 12.4 of Reference 7)	In most cases, for immersion scenarios, conservative or maximum spill rates were used in flood height calculations to determine the potential for floodwater accumulation, time to reach critical flood height, and other relevant factors. For example, the maximum flow that can be delivered by the ruptured pipe or the maximum flow of the pump was used for major flooding scenarios. For non-major flooding scenarios, a maximum flow rate of 2,000 gpm was used.	This modeling approach represents a conservatism in the analysis.	While using maximum spill rates may be conservative, this approach is consistent with industry practice and therefore this uncertainty topic screens based on Criterion #3.	This item does not represent a key source of uncertainty for the RICT calculations.

Table E9-2: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Internal Flood PRA)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT Process
2	Critical Flood Height (Section 12.4 of Reference 7)	This uncertainty topic covers potential issues with critical flood height and the set of target PRA components damaged by flood water immersion. For most flooding scenarios, the critical flood height was assumed to be the lowest elevation of the target PRA components located in the flood area. This could result in conservative HEPs for the flood mitigation actions. The conservatism is a result of the critical flood height assumption allowing less time available to execute the action than if a higher critical flood height was used. The time window thus obtained was used in the evaluation of the HEP for operator flood mitigation/isolation. Given failure of the operators to successfully complete the mitigation action within this time window, all of the target PRA components located in the flood area were assumed to be failed by the flood water. This is a conservative treatment because many of the target PRA components are higher than the critical flood height and likely would not be damaged.	This modeling approach is a significant conservatism in the analysis and could potentially impact risk application results. The critical flood height assumptions result in conservative HEPs or overly conservative flood damage sets.	Overly conservative flood target damage sets may cause failure of more equipment, than if a more realistic approach were utilized. This could result in masking the delta risk of flood mitigating components.	**It is recommended that a sensitivity case be added for the RICT calculations.**

Table E9-3: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Fire PRA)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT
1	Fire Component Selection (Item 2.05 of Table A-2 of Reference 8)	Since there is typically no information concerning the routing of main feedwater (MFW) or instrument air (IA) cables, it is commonly assumed MFW and IA are unavailable for accident mitigation, unless an exclusion analysis is performed to demonstrate that main feedwater, instrument air and/or their support systems are not impacted by a specific fire. Reference 8 recommends a sensitivity analysis on total CDF/LERF assuming IA and MFW are available. This sensitivity was addressed in the "No Tier 3 and UNL sensitivity" found in section 6.2 of the Fire Risk Quantification notebook (Reference 12)	PRA credited components for which cable routing information was not provided represent a source of uncertainty (conservatism) in that are globally failed in the fire PRA could be assumed failed unnecessarily.	The current approach employed is an industry method for concentrating detailed PRA analysis on high-risk items. Low risk and low significance items are given more conservative treatments since they are of lower importance. Therefore, per Criterion #3, this is not a key source of uncertainty for the RICT application since accepted industry methods are applied. If additional equipment is utilized or removed from the RICT case model, then the selection of components could impact the RICT model. However, by not crediting MFW or IA in the base fire model and the RICT case fire model, there would be no impact on delta risk metrics. Based on this, the uncertainty screens on Criterion #1.	This item does not represent a key source of uncertainty for the RICT calculations.
2	Quantitative Screening Item 7.01 of Table A-2 of Reference 8)	Instrument air and service air systems were not credited in the Fire PRA and are assumed to be failed. Instrument air is not an Appendix R system and does not have associated cable tracing to support it.	See Item 1 above.	See Item 1 above.	See Item 1 above.
3	Ignition source screening (Item 8.01 of Table A-2 of Reference 8)	Motor Control Centers (MCCs) are screened as non-propagating ignition sources provided they have adequate breaker protection and are not vented.	No impact; More recent documentation provided in the Fire Modeling documentation (Reference15) shows that fire propagation outside of MCCs was accounted for in the Davis Besse fire	N/A; no impact on risk applications.	This item does not represent a key source of uncertainty for the RICT application.

Table E9-3: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Fire PRA)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT
			PRA model.		
4	Assumed Target Location (Item 8.15 of Table A-2 of Reference 8)	Targets assigned as "Tier 3" are not located in the field or on drawings and are assumed to be damaged in all fire scenarios that are capable of impacting targets beyond the ignition source itself.	See Item 1 above.	See Item 1 above.	See Item 1 above.
5	Circuit Failure Likelihood Analysis (Item 10.06 of Table A-2 of Reference 8)	For multi-conductor (M/C) cables, the probability of both intra-cable hot shorts (internal) and inter-cable hot shorts (external) will be considered. Suppose the M/C cable is in a cable tray or conduit, and both a single conductor (1/C) and an M/C are also located in the same raceway. In that case, the more conservative (higher) probability value for a hot short will be considered. For example, the inter-cable hot short probability for an M/C "target cable" with a control power transformer (CPT) in a tray that has both 1/C and M/C cables will be assumed to be 0.1. Sensitivities in sections 6.1 and 6.2 of the Fire Risk Quantification notebook (Reference 12) address the uncertainty topic of spurious operation probabilities.	Circuit failure mode likelihood analysis was applied to components where spurious operation was expected to be a significant contributor to total risk. Selecting the more conservative hot short probability approach could introduce conservatism into the analysis.	Selecting the more conservative hot short probability could introduce conservatism in the Circuit Failure Likelihood analysis. But it is standard industry practice to choose the more conservative option when dealing with unknowns. Therefore, according to Criterion #3, this is not a key source of uncertainty for RICT, as accepted industry methods are applied.	This item does not represent a key source of uncertainty for the RICT application.
6	Circuit Failure Likelihood Analysis (Item 10.07 of Table A-2 of Reference 8)	Suppose power cables are in the same compartment as the "target cable" control cable. In that case, it is assumed that the power cable is not affected by the fire, in consideration of a hot short resulting in a spurious operation	Credit for the power loss could reduce the number of spurious operations that need to be considered for a given scenario. Therefore, the assumption that the hot	Assuming hot shorts always preceded power loss could make the fire PRA conservative. The approach aligns with standard industry practice, which typically does not credit failures. Therefore, according to Criterion #3, this is not a	This item does not represent a key source of uncertainty for the RICT application.

Table E9-3: Potential Impacts of DBNPS Key Assumptions and Sources of Uncertainty (Fire PRA)

#	Assumption/ Uncertainty	Summary of Assumption/Uncertainty	Generic Impact on Risk Applications	Evaluation of RICT Impact	Suggested Approach for RICT
		(i.e., the hot short is assumed to occur before power is lost). Sensitivities in sections 6.1 and 6.2 of the Fire Risk Quantification notebook (Reference 12) address the uncertainty topic of spurious operation probabilities.	shorts always occur before power could cause the fire PRA results to be conservative.	key source of uncertainty for the RICT calculations, as accepted industry methods are applied.	
7	Breaker Coordination (Item 5.50 of Table A-2 of Reference 8)	Breaker coordination of Non-Essential MCCs are not modeled; assumed included in component data and IE frequencies (Reference 8).	No impact; More recent documentation provided in the Fire PRA Response Model notebook (Reference 16) states that breaker coordination impacts have since been modeled in the fire PRA.	N/A; no impact on risk applications.	This item does not represent a key source of uncertainty for the RICT application.

Table E9-4: Key Sources of Uncertainty Assessment for Risk-Informed Applications

Topic (TR-1016737, Table A-3)	Discussion of Issue	DBNPS Evaluation of RICT Impact
1. Treatment of boron dilution events.	The treatment of boron dilution in PWRs may vary significantly in both the deterministic models and the quantitative probabilistic models. Some applications of the model might be driven by the specific treatment chosen. Typically, these events proceed slowly and are prevented before they become a potential core damage scenario.	Boron dilution is addressed as part of Loss of RCS makeup initiating event (Reference 17). Because it is included as part of a non-support system initiating event, this uncertainty can be screened from further consideration per Criterion #1.
2. Selection of prior distributions when carrying out a Bayesian analysis of data.	Generally, the peer review should be sufficient to ensure that reasonable prior values are employed. However, some applications may be driven by uncertainty in the selection of prior distributions used for either initiating event frequency development or component failure data development.	NUREG/CR-6928 is used for most of the prior data. A review of Bayesian updated data is performed in the Data notebook (Reference 18) to ensure consistency between prior and plant-specific data. There are no open peer review Facts & Observations (F&Os) about this issue (Reference 19). Per Criterion #3, this item does not represent a key source of uncertainty for the RICT application.
3. Treatment of rare and extremely rare events	Generally, the peer review should be sufficient to ensure that reasonable values are employed. The selection of data should be based on confirmation that the database used applies to the plant (e.g., no unique failure modes not considered in the database are active).	Per the Initiating Events PRA Notebook (Reference 17), the frequencies of all identified initiating events were determined using plant specific trip information from licensee event reports (LERs), simplified fault tree failure models of systems, and generic data from various sources. There are no open F&Os pertaining to this issue (Reference 19). Per Criterion # 3, this is standard PRA practice and will not impact the RICT application.
4. Moderator temperature coefficient (MTC)	The peer review should address the basis for the MTC used in the PRA based on plant-specific data. This could be an application-specific uncertainty, however, especially for an application focused on ATWS mitigation.	Currently, no modifications impact the fuel characteristics related to MTC. However, it should be noted that Davis Besse is pursuing an 80-MW uprate, which is scheduled to be completed in 2034. Upon implementation of the associated modifications, this uncertainty will need to be revisited; However, according to Criterion #1, it is not currently a concern for the RICT application.

Table E9-4: Key Sources of Uncertainty Assessment for Risk-Informed Applications

Topic (TR-1016737, Table A-3)	Discussion of Issue	DBNPS Evaluation of RICT Impact
5. Pressurized Thermal Shock (PTS) – PWRs.	Consideration of pressurized thermal shock after the secondary system is depressurized, for example, after a main steam line break, can lead to sources of model uncertainty.	PTS is subsumed into Reactor Vessel Rupture and goes immediately to core damage. This approach is consistent with Industry Practice, and is not a source of model uncertainty. Therefore, per Criterion #3, this potential uncertainty is not of concern the RICT application.
6. Credit for non-standard success paths (e.g., use of alternate injection systems)	In general, the peer review should be adequate to ensure that unreasonable credit is not given for non-standard success paths. However, some applications might be driven by the uncertainty involved in crediting some non-standard success paths.	Non-standard success paths credited in the DBNPS PRA are associated with FLEX mitigation strategies including use of portable equipment. Use of these strategies is proceduralized in EOPs, however, uncertainty associated with credit for FLEX equipment in light of the NRC memorandum dated May 6, 2022 (Reference 20) on the challenges and strategies for incorporating FLEX equipment into the PRA model should be assessed for impacts on the RICT application. **It is recommended that a sensitivity case be added for the RICT application.**
7. CDF and LERF definitions – the PRA standard allows some flexibility in defining these parameters.	Definitions should be clear and justified. The peer review should be adequate to make sure that reasonable definitions are used. However, non-standard definitions could lead to potential key sources of model uncertainty since it impacts timing and response requirements.	DBNPS definitions utilize the industry's PWR "standard" definitions and have been peer reviewed. The Davis Besse definitions for CDF and LERF agree with those in the Standard. There are no open peer-reviewed F&Os (Reference 19) about this issue. Per Criterion #3, this is not a source of uncertainty for the RICT application.

Table E9-4: Key Sources of Uncertainty Assessment for Risk-Informed Applications

Topic (TR-1016737, Table A-3)	Discussion of Issue	DBNPS Evaluation of RICT Impact
8. Large LOCA long term oxidation in BWRs – since BWRs are designed to maintain 2/3 core height for a very large break LOCA, injection by one LPCI pump into the shroud area may maintain the covered core sub-cooled. Cooling of the top 1/3 core for a substantial time is questionable since long term steam cooling effect may not be ensured.	The body of technical work that supports the assumption of a single LPCI pump for applicable plants should be sufficient on a generic basis for the timeframes considered in a PRA. Additionally, alternative success criteria may be employed (e.g., long-term success for LPCI injection also requires the implementation of containment flooding strategies, as would be required per the SAMGs). However, this could be a source of uncertainty specific to the application.	N/A, DBNPS is a PWR.
9. Engineering analyses – separate engineering analyses may use codes or invoke other assumptions that may introduce potential sources of modeling uncertainty.	Table A-1 does include the engineering analyses that should be most important (i.e., room heatup and battery depletion calculations). Additionally, unique plant-specific analyses would be expected to be identified as a plant-specific source of model uncertainty. For other engineering analysis, If the codes or methods are accepted by NRC and industry, then the engineering analysis may meet the consensus approach criteria. If not, then the analysis may be a source of modeling uncertainty.	The ASME/ANS PRA standard (Reference 3) was reviewed for applicable supporting requirements regarding plant specific analyses, codes, engineering analyses, and engineering bases. All supporting requirements were met. There are no open finding level F&Os pertaining to this issue (Reference 19). Per Criterion #3, this is not a source of uncertainty for the RICT application.

Table E9-4: Key Sources of Uncertainty Assessment for Risk-Informed Applications

Topic (TR-1016737, Table A-3)	Discussion of Issue	DBNPS Evaluation of RICT Impact
10. Level control during ATWS in BWRs – difficult to perform, but more importantly, the power level achieved in different situations is uncertain. Power/flow oscillations can occur, and their impact on the core is uncertain.	The peer review should be adequate to ensure that reasonable success criteria are used. There are HRA uncertainties, but these should be subsumed into the general treatment of HEP uncertainties. This could be an application-specific uncertainty, however, especially for an application focused on ATWS mitigation.	N/A, DBNPS is a PWR.
11. Post-LOCA boron precipitation in PWRs – modeled in design basis event thermal hydraulic evaluations, but is not always modeled in PRAs.	Research is underway to clarify this issue further. The treatment of this phenomenon will impact long-term cooling success criteria following larger LOCAs.	Post-LOCA Boron Precipitation Control (BPC) is not included in the PRA. This impacts the significance of Reactor Protection System (RPS) equipment. Because the Fussell-Vesley of the RPS system less than 1E-4 (Reference 5) this is not a significant source of model uncertainty. Also, the assumption of not modeling boron dilution is made based on the level of detail, e.g., modeling simplification where further modeling would reduce the CDF (Reference 5). Per Criterion #1 and #2, this item does not represent a key source of uncertainty for the RICT application.

Table E9-4: Key Sources of Uncertainty Assessment for Risk-Informed Applications

Topic (TR-1016737, Table A-3)	Discussion of Issue	DBNPS Evaluation of RICT Impact
12. Digital instrumentation and control.	Some plants have incorporated digital systems into their designs or to replace existing analog systems. There are model uncertainties associated with modeling digital systems, such as those related to determining the failure modes of these systems and components.	The Electro-Hydraulic Control (EHC) system has been upgraded to a digital system; however, EHC is not explicitly modeled in the PRA, and failures in the system would be considered as part of the reactor/turbine trip initiating event. Based on Criterion #1, this aspect of Digital I&C at DBNPS can be screened from further evaluation. Another digital upgrade at DBNPS is associated with the Emergency Diesel Generator (EDG) Excitation system, specifically the replacement of a motor-operated rheostat (MOR) with a microprocessor-based reference unit for establishing the voltage regulator setpoint. The Evaluation of Reliability of Replacement EDG Excitation System for Davis-Besse Nuclear Power Plant (Reference21) concludes that the microprocessor-based reference unit will not increase the failure probability of the EDG system. Per Criterion #3, this topic does not represent a key source of uncertainty for RICT.
13. Credit for non-safety related equipment in recovery actions	This may involve the use of portable equipment and/or flexible hoses. Some of the equipment may be pre-staged, and the actions may be procedurally directed, but not all of them.	The parts of the model potentially impacted by this uncertainty are those that credit FLEX strategies. Uncertainty associated with FLEX is addressed in Item 6 above. These strategies are proceduralized, and appropriate human error probabilities have been modeled as documented in the human reliability assessment notebook (Reference 13). The uncertainty of HEPs has been addressed separately. No additional uncertainties are judged to exist for crediting non-safety-related equipment in recovery actions. Per Criterion #1, the uncertainty associated with this topic will be addressed as part of the RICT process.

Table E9-4: Key Sources of Uncertainty Assessment for Risk-Informed Applications

Topic (TR-1016737, Table A-3)	Discussion of Issue	DBNPS Evaluation of RICT Impact
14. Passive system degradation mechanisms – aging of active components is incorporated into the periodic data analysis updates but passive system reliability is generally not accounted for.	Plant-specific data is based on recent, applicable data representative of current and future plant operations. Otherwise, it relies on generic data sources. This issue may be significant for applications of the model that extrapolate into longer time intervals and assessments of pipe failure frequency.	Flood initiator frequencies are based on plant-specific estimates of pipe lengths and on the predominant use of industry-generic data for pipe failure frequencies specific to different piping categories, as outlined in the EPRI methodology. (Reference 7) The use of generic flood frequencies, updated to account for plant-specific operating experience, design, and operating practices, in conjunction with plant-specific estimates of pipe lengths, is suitable for representing the flood frequencies at the site. Per Criterion #3, this is considered an industry-consensus and good-practice approach and does not represent a source of uncertainty for RICT.
15. Water hammer impacts on system performance.	A water hammer event can cause significant stress in pipes and components. The analysis of pipe and component failures resulting from a water hammer event is not generally available. The incorporation or lack of incorporation of the impact of water hammer events may be relevant in specific model applications.	Per the peer review documentation (Reference 19), water hammer is addressed in the internal flooding analysis. This topic does not represent a key source of uncertainty for RICT. Per Criterion #3, this topic does not represent a key source of uncertainty for RICT.
16. Selection of components in a common cause group.	The choice of common cause groups or the lack of identification of certain common cause groups could influence specific applications of the model.	The PRA model CCF group selections have been peer-reviewed, and there are no open findings about this topic (Reference 19). As new components are added to the model, the CCF aspect is considered and included if required. Per Criterion #3, this topic does not represent a key source of uncertainty for RICT.

Table E9-4: Key Sources of Uncertainty Assessment for Risk-Informed Applications

Topic (TR-1016737, Table A-3)	Discussion of Issue	DBNPS Evaluation of RICT Impact
17. The capability of the battery charger to start and carry loads if the battery is unavailable.	Credit for the charger to start and carry several loads simultaneously may not be appropriate, depending on the charger's rating compared to the corresponding battery's rating.	N/A. The plant DC System notebook (Reference 22) states that DBNPS, a battery charger, is normally aligned to charge each station battery and provide for steady-state DC loads. (i.e., Safety-related battery chargers are capable of both providing all loads and charging the battery simultaneously). Therefore, this topic does not represent a key source of uncertainty for RICT.
18. Standby failure rate model.	The selection of the standby failure rate model could be relevant for specific applications of the model (specifically for surveillance frequency change evaluations).	N/A. This topic is specifically applicable to surveillance frequency change evaluations and, therefore, does not represent a key source of uncertainty for RICT.

3.0 Assessment of Translation (Real Time Risk Model) Uncertainty Impacts

Incorporation of the baseline PRA models into the CRM model used for RICT Program calculations may introduce new sources of model uncertainty. Table E9-5 provides a description of the relevant model changes and dispositions of whether any of the changes made represent possible new sources of model uncertainty that must be addressed. Refer to Enclosure 8 for additional discussion on the CRM model.

Table E9-5: Assessment of Translation Uncertainty Impacts

CRM Model Change and Assumption	Part of Model Affected	Impact on Model	Disposition
PRA model logic structure may be optimized to increase solution speed.	Fault tree logic model structure, affecting internal events PRAs	The model, if restructured, will be logically equivalent and produce results comparable to the base PRA logic model.	Since the restructured model will produce comparable numerical results, this is not a source of uncertainty for the RICT program.
Incorporation of seismic risk bias to support RICT Program risk calculations. Conservative values for the seismic delta CDF and LERF are applicable.	Calculation of RICT and risk management action time (RMAT) within the CRM model	The addition of conservative impacts for seismic events has no impact on base PRA or CRM model. Impact is reflected in calculation of all RICTs and RMATs.	Since this is a conservative approach to addressing seismic risk in the RICT Program, it does not introduce translation uncertainty, and the RICT Program's calculations are unaffected. Therefore, no mandatory Risk Management Actions (RMAs) are required.

4.0 References

1. Letter from the NRC to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)," dated May 17, 2007 (ADAMS Accession No. ML071200238).

2. EPRI Technical Report TR-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," dated December 2008.

3. ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RA-S-2008, *Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications*, February 2, 2009.

4. NUREG-1855, *Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking*, Revision 1, March 2017.

5. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station PRA Notebook, 08-01: Quantification,* Revision 7.

6. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station PRA Notebook, 07-01: Level 2*, Revision 7.

7. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station Internal Flooding PRA Notebook, 09-12: Quantitative Analysis (Task 10),* Revision 7.

8. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station PRA Notebook, 10-06: Fire Sensitivity and Uncertainty,* Revision 7.

9. ENERCON Calculation: VISTRADB-004-REPT-003, Rev. 0, DBNPS Impact of Model Uncertainty on the RICT Process.

10. NUREG/CR-6850 (EPRI 1011989), *Fire PRA Methodology for Nuclear Power Facilities Volume 2: Detailed Methodology*, September 2005.

11. ENERCON Calculation: VISTRADB-00004-CALC-001, Rev. 0, Evaluation of RICT Durations for Davis-Besse Nuclear Power Station.

12. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station PRA Notebook, 10-05: Fire Risk Quantification*, Revision 7.

13. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station PRA Notebook, 05-01: Human Reliability Analysis Notebook, Revision 7.*

14. EPRI 1021081, *Establishing Minimum Acceptable Values for Probabilities of Human Failure Events*, Oct. 2010.

15. C-FP-013.10-009, Fire Modeling, Revision 3

16. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station PRA Notebook, 10-02: Fire PRA Response Model, Revision 7.*

17. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station PRA Notebook, 02-01: Initiating Events Volume 1*, Revision 7.

18. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station PRA Notebook, 06-01: Data Volume 1 Generic & Plant-Specific Failure Rates*, Revision 7.

19. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station PRA Notebook, 01-01: PRA Roadmap and Peer Reviews*, Revision 7.

20. U.S. NRC memorandum, “Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Risk Assessments,” dated May 6, 2022 (ADAMS Accession No. ML22014A084).

21. MPR-2838, Evaluation of Reliability of Replacement EDG Excitation System for Davis-Besse Nuclear Power Plant, Revision 1.

22. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station PRA Notebook, 04-11: 125/250V DC and 120V Instrumentation AC*, Revision 7.

23. PRA-DB1-AL-R07, *Davis Besse Nuclear Power Station PRA Notebook, 04-21: Aux Building Non-Rad Waste Area HVAC, Revision 7.*

24. EPRI 1009652, “Guideline for the Treatment of Uncertainty in Risk-Informed Applications,” Electric Power Research Institute, December 2004.

Enclosure 10

Davis-Besse Nuclear Power Station

License Amendment Request

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b”

Program Implementation

Program Implementation

1.0 Introduction

Section 4.0, Item 11 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A (Reference 2) requires that the license amendment requires (LAR) provide a description of the implementing programs and procedures regarding plant staff responsibilities for the Risk Managed Technical Specifications (RMTS) implementation, and specifically discuss the decision process for risk management action (RMA) implementation during a Risk-Informed Completion Time (RICT).

This enclosure provides a description of the implementing programs and procedures regarding the plant staff responsibilities for the RICT program, including training of plant personnel, and specifically discusses the decision process for RMA implementation during extended Completion Times (CT). Enclosure 12 provides additional details regarding the process for developing and implementing RMAs.

2.0 RICT Program and Procedures

Vistra will develop a program description and implementing procedure for the RICT program. The program description will establish the management responsibilities and general requirements for risk management, training, implementation, and monitoring of the RICT program. Procedures will provide specific responsibilities, limitations, and instructions for implementing the RICT program. The program description and implementing procedures will incorporate the programmatic requirements for RMTS in accordance with NEI 06-09-A. The program will be integrated with the online work control process. The work control process currently identifies the need to enter a Limiting Condition for Operation (LCO) action statement as part of the planning process and will additionally identify whether the provisions of the RICT program are requirements for the planned work. The risk thresholds associated with the 10 CFR 50.65(a)(4) performance monitoring provisions and Mitigating System Performance Index (MSPI) thresholds will assist in controlling the amount of risk expended in use of the RICT program.

The Operations Department (licensed operators) is responsible for compliance with the Technical Specifications (TS) and will be responsible for the implementation of the RICTs and RMAs. Entry into the RICT program will require management approval prior to pre-planned activities and as soon as practicable following emergent conditions.

The procedures for the RICT program will address the following attributes consistent with NEI 06-09-A:

- Plant management positions with authority to approve entry into the RICT program.
- Important definitions related to the RICT program.
- Departmental and position responsibilities for activities in the RICT program.
- Plant conditions for which the RICT program is applicable.
- Implementation of the RICT program 30-day back-stop CT limit.
- Limitations on implementing RICTs under voluntary and emergent conditions.

- Use of the Configuration Risk Management Program (CRMP) tool.
- Implementation of the RICT and risk management action time (RMAT) within 12 hours or within the most limiting front-stop CT after a plant configuration change.
- Requirement to identify and implement RMAs when the RMAT is exceeded or is anticipated to be exceeded, and to consider common cause failure potential in emergent RICTs.
- Guidance on the use of RMAs including the conditions under which they may be credited in RICT calculations.
- Guidance on crediting PRA functionality.
- Conditions for exiting a RICT.
- Requirements for training on the RICT program.
- Documentation requirements related to individual RICT evaluations, implementation of extended CTs, and accumulated annual risk.

3.0 RICT Program Training

Training will be carried out in accordance with Vistra training procedures and processes that utilize the Systematic Approach to Training. These procedures were written based on the Institute of Nuclear Power Operations (INPO) Accreditation requirements, as developed and maintained by the National Academy for Nuclear Training.

Participation

Departments that will receive training appropriate to their level of program responsibilities include:

- Operations
- Operations Training
- Work Management
- Outage Management
- Planning and Scheduling Personnel
- Work Week Managers
- Regulatory Assurance
- Maintenance
- Engineering
- PRA Staff
- Other Selected Management

Scope of Training

For those individuals who will be directly involved in the implementation of the RICT program, the training will include the following topics:

- Specific training on the revised TS
- Record keeping requirements
- Case studies

- Hands-on experience with the CRMP tool for calculation RMAT and RICT
- Identifying appropriate RMAs
- Determining PRA functionality
- Common cause failure considerations
- Other detailed programmatic requirements of the RICT program

For management positions with authority to approve entry into the RICT program, as well as supervisors, managers, and other personnel who will closely support RICT implementation or need a general awareness, the training will provide a broad understanding of purpose, concepts and limitations of the program.

4.0 References

1. Letter from the NRC to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)," dated May 17, 2007 (ADAMS Accession No. ML071200238)
2. NEI Topical Report NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, dated October 2012 (ADAMS Accession No. ML12286A322)

Enclosure 11

Davis-Besse Nuclear Power Station

License Amendment Request

Revise Technical Specifications to Adopt RISK Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b"

Monitoring Program

Monitoring Program

1.0 Introduction

Section 4.0, Item 12 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A (Reference 2) requires that the license amendment request (LAR) provide a description of the implementation and monitoring program as described in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", Revision 1 (Reference 3), and NEI 06-09-A. Note that Revision 2 of RG 1.174 (Reference 4) was issued by the NRC in May 2011 and made editorial changes to the applicable section of RG 1.174 referenced in Section 4.0, Item 12 of the NRC Safety Evaluation. Also, Revision 3 of RG 1.174 (Reference 5) was issued by the NRC in January 2018 and the relevant guidance regarding the implementation and monitoring program remain substantially the same.

This enclosure provides a description of the process applied to monitor the cumulative risk impact of implementation of the Risk-Informed Completion Time (RICT) program, specifically the calculation of cumulative risk of extended Completion Times (CTs). Calculation of the cumulative risk for the RICT program is discussed in Step 14 of Section 2.3.1 and Step 7.1 of Section 2.3.2 of NEI 06-09-A (Reference 2). General guidance for a performance monitoring program for risk-informed applications is discussed in Element 3 of RG 1.174, Revision 3 (Reference 5)

2.0 Description of Monitoring Program

The RICT program will require calculation of cumulative risk impact at least every refueling cycle, not to exceed 24 months, consistent with the guidance in NEI 06-09-A (Reference 2). For the assessment period under evaluation, data will be collected for the risk increase associated with each application of an extended CT for both core damage frequency (CDF) and large early release frequency (LERF). The total risk impact will be calculated by summing all risk associated with each RICT application. This summation is the change in CDF or LERF above the zero maintenance baseline levels during the period of operation in the extended CT (i.e., beyond the front-stop CT). The change in risk will be converted to average annual values and documented every fuel cycle.

The total average annual change in risk for extended CTs will be compared to the guidance of RG 1.174, Revision 3 (Reference 5), Figures 4 and 5, acceptance guidelines for CDF and LERF, respectively. If the actual annual risk increase is acceptable (i.e., not in Region I of Figures 4 and 5 of RG 1.174), then RICT program implementation is acceptable for the assessment period. Otherwise, further assessment of the cause of exceeding the acceptance guidelines of RG 1.174 and implementation of any necessary corrective actions to endure future plant operation is within the guidelines will be conducted under the Corrective Action Program.

The evaluation of the cumulative risk will also identify areas for consideration, such as:

- RICT applications that dominated and incurred a large portion of the risk increase
- Risk contributions from planned versus emergent RICT applications
- Risk Management Actions (RMAs) implemented but not credited in the risk calculations

- Risk impact from applying RICT to avoid multiple shorter duration outages

Based on a review of the considerations above, corrective actions will be developed and implemented as appropriate. These actions may include:

- Administrative restrictions of the use of RICTs for specific high-risk configurations
- Additional RMAs for specific configurations
- Rescheduling planned maintenance activities
- Deferring planned maintenance to shutdown conditions
- Use of temporary equipment to replace out-of-service systems, structures, or components (SSCs)
- Plant modifications to reduce the risk impact of future planned maintenance configurations

In addition to impacting cumulative risk, the implementation of the RICT program may potentially impact the unavailability of SSCs. The Maintenance Rule (MR) monitoring programs under 10 CFR 50.65 provide for evaluation and disposition of unavailability impacts that may be incurred from implementation of the RICT program. The MR program can be utilized for monitoring the performance of those SSCs that are in the scope of the RICT program and the scope of the MR. The Davis-Besse Nuclear Power Station is based on NUMARC 93-01 (Reference 6).

The monitoring program of the MR, along with the specific assessment of cumulative risk impact described above, serve as the RICT program implementation and monitoring program as described in Element 3 of RG 1.174 (Reference 5) and NEI 06-09-A (Reference 2).

3.0 References

1. Letter from the NRC to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)," dated May 17, 2007 (ADAMS Accession No. ML071200238)
2. NEI Topical Report NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, dated October 2012 (ADAMS Accession No. ML12286A322)
3. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, dated November 2002 (ADAMS Accession No. ML023240437)
4. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, dated May 2011 (ADAMS Accession No. ML100910006)
5. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, dated November 2002January 2018 (ADAMS Accession No. ML17317A256)
6. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4A, dated April 2011 (ADAMS Accession No. ML11116A198)

Enclosure 12

Davis-Besse Nuclear Power Station

License Amendment Request

Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b"

Risk Management Action Examples

Risk Management Action Examples

1.0 Introduction

This enclosure describes the process for identification and implementation of Risk Management Actions (RMAs) applicable during extended Completion Times (CTs) and provides examples of how the process would be implemented. Vistra procedures for planning and scheduling maintenance activities will govern RMAs. The procedures will provide guidance for the determination and implementation of RMAs when entering the Risk-Informed Completion Time (RICT) program consistent with the guidance provided in NEI 06-09-A (Reference 1).

2.0 Responsibilities

For planned entries into the RICT program, Work Management is responsible for developing the RMAs with assistance from Operations and the PRA Staff. For emergent entry into extended CTs, Operations is responsible for developing the RMAs, but may seek assistance from the PRA Staff or Work Management. Operations is responsible for approval and implementation of all RMAs (for planned and emergent conditions).

3.0 Procedural Guidance

For planned maintenance activities, implementation of RMAs will be required if it is anticipated that the risk management action time (RMAT) will be exceeded. For emergent activities, RMAs must be implemented if the RMAT is reached. Also, if an emergent event occurs requiring recalculation of an RMAT already in place, the procedure will require a reevaluation of the existing RMAs for the new plant configuration to determine if additional RMAs are appropriate. These requirements of the RICT program are consistent with the guidance of NEI 06-09-A.

For emergent entry into a RICT, if the extent of condition is not known, RMAs related to the success of redundant and diverse structures, systems, and components (SSCs) will be developed and implemented to address the potential for common cause failure modes. These RMAs will focus on reducing the likelihood of initiating events that rely on the affected function as well as protecting the in-service equipment that performs the redundant and diverse functions.

RMAs will be implemented no later than the time at which an incremental core damage probability (ICDP) of 1E-06 is reached, or no later than the time when an incremental large early release probability (ILERP) of 1E-07 is reached. If, as the result of an emergent condition, the instantaneous core damage frequency (ICDF) or the instantaneous large early release frequency (ILERF) exceeds 1E-03 per year or 1E-04 per year, respectively, RMAs are also required to be implemented. These requirements are consistent with the guidelines of NEI 06-09-A.

By determining which initiators, fire compartments, or components are most important from a CDF or LERF perspective for a specific plant configuration, RMAs may be created to protect these components or increase awareness as it relates to their importance. Similarly, knowledge of the initiating event or sequence contribution related to the configuration-specific CDF or LERF allows development of RMAs that enhance the capability to mitigate such events. The guidance in

NUREG-1855 (Reference 2) and EPRI Technical Update 1026511 (Reference 3) will be used in examining PRA results for significant contributors for the configuration, to aid in identifying appropriate compensatory measures (e.g., related to risk-significant systems that may provide diverse protection, important support systems, important human actions).

If the planned activity or emergent condition includes an SSC that is identified to impact fire PRA, as identified in the Configuration Risk Management Program (CRMP), fire PRA specific RMAs associated with that SSC will be implemented per procedure. Approved equipment specific RMAs for risk significant SSCs within the scope of the RICT program will be contained in the procedure. Common cause RMAs will also be considered for emergent conditions where the extent of condition cannot rule out the potential for common cause failures.

It is possible to credit RMAs in RICT calculations, to the extent the associated plant equipment and operator actions are modeled in the PRA; however, such quantification of RMAs is neither required nor expected by NEI 06-09-A. Nonetheless, if RMAs will be credited in RICT calculations, procedure instructions will be consistent with the guidance in NEI 06-09-A.

NEI 06-09-A classifies RMAs into the three categories described below:

1. Actions to increase risk awareness and control.
    - Conduct shift briefings
    - Conduct pre-job briefings
    - Require the presence of the system engineer or other expertise related to the activity
    - Receive management approval of the proposed activity
    - Prioritize the restoration of out-of-service components
    - Identify and protect important in-service components
2. Actions to reduce the duration of maintenance activities
    - Pre-stage parts and materials
    - Perform walk-downs of the system tag-outs and key equipment prior to beginning the work
    - Develop critical activity procedures for risk-significant configurations, including identification of the associated risk and contingency plans for approaching/exceeding the RICT
    - Conduct training on mock-ups to familiarize maintenance personnel prior to the work
    - Perform the activity around the clock rather than “day-shift only”
    - Establish contingency plans to restore key out-of-service equipment rapidly if needed

3. Actions to minimize the magnitude of the risk increase.

    - Suspend or minimize activities on or in the vicinity of redundant systems
    - Suspend or minimize activities on other systems that adversely affect the CDF or LERF
    - Suspend or minimize activities on systems that may cause a trip or transient to minimize the likelihood of an initiating event that the out-of-service component is designed to mitigate
    - Use temporary equipment to provide backup power, ventilation, etc.
    - Reschedule other risk-significant activities
    - Expedite equipment return to service to reduce risk levels
    - Protect equipment that provides alternate diverse success paths for performing the safety function of the out-of-service SSC

Determining RMAs involves the use of both qualitative and quantitative considerations for the specific plant configuration and the practical means available to manage risk. A graded approach is used to identify the scope of RMAs that are appropriate for the plant configuration. Procedural guidance for the development of RMAs in support of the RICT program builds off other processes, such as actions taken under the 10 CFR 50.65(a)(4) program and the protected equipment program. Additionally, common cause RMAs may be developed to address the potential impact of common cause failure modes and may be performed in conjunction with the Davis-Besse Nuclear Power Station Safety Function Determination Program (SFDP), per Technical Specification (TS) 5.5.14.

Vistra procedures will provide general guidance for developing RMAs, such as:

- Consideration of rescheduling maintenance to reduce risk
- Discussion of RICT in pre-job briefs
- Consideration of proactive return-to-service of other equipment
- Efficient execution of the maintenance

In addition to RMAs developed qualitatively, RMAs are developed based on the CRMP to identify configuration-specific RMA candidates to manage the risk associated with internal events, internal flooding, and fire events. These actions include:

- Identification of important equipment or divisions for protection
- Identification of important operator actions for crew briefings or just-in-time training
- Identification of key fire initiators and fire zones for RMAs

- Identification of dominant initiating events and actions to minimize their occurrence
- Consideration of insights from dominant PRA model cutsets

Common cause RMAs may also be developed to ensure availability of redundant SSCs, to ensure availability of diverse systems, to reduce the likelihood of initiating events that the out-of-service components are designed to mitigate, and to prepare plant personnel to respond to additional failures. Common cause RMAs are developed by considering the impact of loss of function for the affected SSCs.

Examples of common cause RMAs include:

- Performance of non-intrusive inspections on alternate divisions
- More frequent monitoring for running or standby components
- Expansion of component monitoring
- Deferring maintenance and testing activities that could generate or increase the frequency of an initiating event which would require operation of potentially affected SSCs
- Readiness of operators and maintenance to respond to additional failures
- Shift briefs or standing orders that focus on initiating event response or loss of potentially affected SSCs
- Engineering evaluation of a failed component to determine the most likely cause and to evaluate the susceptibility of other plant equipment

Vistra procedures will require that for emergent conditions where the extent of condition is not performed prior to exceeding the RMAT or the extent of condition cannot rule out the potential for common cause failure, common cause RMAs are expected to be implemented. These RMAs can include the pre-identified, general RMAs as discussed above, as well as alternative common cause RMAs for the specific configuration.

4.0 **Examples**

Multiple example RMAs that may be considered during a RICT program entry to reduce the risk impact and ensure adequate defense-in-depth are provided below. Specific examples are given for the inoperability of a Diesel Generator (DG), an offsite electrical power circuit, an offsite circuit with a DG, a safety-related 125V DC battery, an electrical distribution subsystem, and a low-pressure Emergency Core Cooling System (ECCS) injection/spray subsystem.

4.1 DG Inoperable

For TS 3.8.1.B, one required DG inoperable, RMAs may include:

1) Actions to increase risk awareness and control.

- Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established. Specific focus areas may be to review emergency or abnormal operating procedures associated with loss of offsite power (LOOP) and station blackout (SBO) events. This may include a review of alternate electrical alignments, considering the inoperable DG, that may be needed during a LOOP or SBO event.

- Perform walkdowns of the remaining operable DGs to validate their standby/readiness condition.

- Communicate the configuration to the transmission system operator so that any planned activities with the potential to cause a grid disturbance can be closely coordinated or deferred.

- Minimize the accumulation of transient combustibles in accordance with the station fire protection program; this would include a heightened sense of awareness on the fire zones that have become more important due to the inoperable DG.

- Challenge any plans to conduct hot work inside the power block or in the switchyard and particularly in the fire zones that have become more important due to the configuration. There should be a bias for not performing hot work during the extended CT (unless it’s directly related to the inoperable DG).

2) Actions to reduce the duration of maintenance activities.

- For preplanned RICT entry, use an equipment outage schedule that identifies and plans all needed resources and provides logic ties between critical activities.

- Confirm parts availability prior to entry into a preplanned RICT.

- Designate additional resources to improve efficiency (e.g., designate a resource to be a parts or tools expediter to maximize wrench time).

- Walkdown systems prior to beginning the work and stage equipment (hoses, fittings, tools, etc.) needed to conduct the tagging and the maintenance.

- Anticipate and be prepared to restore the DG as the maintenance concludes (be prepared to refill lube oil systems, fill and vent coolant water, rack breakers, etc.).

- Ensure the correct post-maintenance DG testing is planned and is ready to be implemented following the maintenance (operators are briefed, procedures are ready, etc.).

- Procure the DG vendor’s services and have them on site to support the equipment outage, if needed.

3) Actions to minimize the magnitude of the risk increase.

- Verify the availability of required offsite circuits every eight hours.

- Ensure safety functions are maintained by continually assessing the status of required features that are redundant to those systems supported by the inoperable DG.
- For emergent RICT entry, determine if the remaining operable DGs are impacted by a common cause failure mode.
- Proactively implement RMAs during times of high grid stress conditions, such as during high demand conditions.
- Evaluate weather conditions for threats to the reliability of offsite power supplies.
- Defer elective maintenance in the switchyard and on the station's electrical distribution systems.
- Defer maintenance or testing that may impact the reliability of operable DGs and associated support equipment.
- Implement 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected DG.
- Implement the protected equipment program for the inoperable DG (protect operable DGs and other important equipment identified in the CRMP).
- Maintain detection, suppression, and fire zone barriers intact and minimize transient combustibles for those fire areas identified as being significant for the configuration.
- Curtail non-essential electrical switching operations to minimize the potential for deenergizing electrical buses.

4.2 Offsite Circuit Inoperable

For TS 3.8.1.A, one required offsite circuit inoperable, RMAs may include:

1) Actions to increase risk awareness and control.

   - Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established. Specific focus areas would be to review emergency or abnormal operating procedures associated with LOOP and SBO events. This may include a review of alternate switchyard alignments considering the inoperable offsite source.
   - Limit access (e.g., vehicle traffic) and other activities in the switchyard; perform periodic walkdowns of the switchyard to validate no unauthorized activities are in progress.
   - Routinely communicate with the transmission system operator so that any planned activities with the potential to cause a grid disturbance can be closely coordinated or deferred.
   - Minimize the accumulation of transient combustibles in accordance with the station fire protection program; this would include a heightened sense of awareness on

the fire zones that have become more important due to the plant configuration (e.g., onsite power sources will become more important).

- Challenge any plans to conduct hot work inside the power block or in the switchyard and particularly in the fire zones that have become more important due to the configuration. There should be a bias for not performing hot work during the extended CT (unless it's directly related to the inoperable offsite circuit).

2) Actions to reduce the duration of maintenance activities.

- For preplanned RICT entry, use an equipment outage schedule that identifies and plans all needed resources and provides logic ties between critical activities.
- Confirm parts availability prior to entry into a preplanned RICT.
- Implement rigorous communications and coordination between site operations and transmission grid operators to ensure the activities rendering the offsite circuit inoperable are progressing as expected and if not, additional resources are applied to reduce the duration of the activity.

3) Actions to minimize the magnitude of the risk increase.

- Verify the availability of the remaining offsite circuit every eight hours.
- Proactively consider additional RMAs during times of high grid stress conditions, such as during high demand conditions.
- Evaluate weather conditions for threats to the reliability of the remaining offsite power source.
- Defer elective maintenance in the switchyard and on the station's electrical distribution systems.
- Defer maintenance or testing that may impact the reliability of DGs and associated support equipment.
- Implement 10 CFR 50.65(a)(4) fire-specific RMAs associated with the offsite source such as maintaining detection, suppression, and fire zone barriers intact and minimizing transient combustibles for those fire areas identified as being significant for the configuration.
- Implement the protected equipment program for the inoperable offsite source. This may include limiting access to the switchyard (e.g., locked gates), barricading and/or posting transformer areas, and/or posting rooms containing intermediate voltage electrical buses.
- Curtail non-essential electrical switching operations to prevent losing the remaining offsite source and to prevent challenging the availability of onsite electrical power.

4.3 Offsite Circuit Inoperable with an Inoperable DG

For TS 3.8.1.D, one required offsite circuit inoperable and one required DG inoperable, RMAs may include:

1) Actions to increase risk awareness and control.

- Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established. Specific focus areas would be to review emergency or abnormal operating procedures associated with LOOP and SBO events. This may include a review of alternate electrical alignments.

- Limit access (e.g., vehicle traffic) and other activities in the switchyard; perform periodic walkdowns of the switchyard to validate no unauthorized activities are in progress.

- Communicate with the transmission system operator so that any planned activities with the potential to cause a grid disturbance can be closely coordinated or deferred. In this condition there should be a bias towards deferral.

- Minimize the accumulation of transient combustibles in accordance with the station fire protection program; this would include a heightened sense of awareness on the fire zones that have become more important due to the plant configuration.

- Perform walkdowns of the remaining operable DGs to validate their standby/readiness condition.

- Challenge any plans to conduct hot work inside the power block or in the switchyard and particularly in the fire zones that have become more important due to the configuration. There should be a bias for not performing hot work during the extended CT (unless it’s directly related to the inoperable offsite circuit or the inoperable DG).

- Establish the outage control center to provide oversight and coordinate return-to-service activities.

2) Actions to reduce the duration of maintenance activities.

- Implement rigorous communications and coordination between site operations and transmission grid operators to ensure the activities rendering the offsite circuit inoperable are progressing as expected and if not, additional resources are applied to reduce the duration of the activity.

- Engage senior managers and executives to ensure the transmission operators are utilizing the necessary resources with an extreme sense of urgency as it relates to restoring the offsite circuit to an operable status (considering the concurrent DG inoperability).

- Anticipate and be prepared to restore the DG as the maintenance concludes (be prepared to refill lube oil systems, fill and vent coolant water, rack breakers, etc.).
- Ensure the correct post-maintenance DG testing is planned and is ready to be implemented following the maintenance (operators are briefed, procedures are ready, etc.).
- Procure the DG vendor's services and have them on site to support the equipment outage, if needed.

3) Actions to minimize the magnitude of the risk increase.

- Verify the availability of the remaining offsite circuit every eight hours.
- Ensure safety functions are maintained by continually assessing the status of required features that are redundant to those systems supported by the inoperable DG.
- Determine if the remaining operable DGs are impacted by a common cause failure mode.
- Implement additional RMAs during times of high grid stress conditions, such as during high demand conditions.
- Evaluate weather conditions for threats to the reliability of the remaining offsite power supplies.
- Allow only mission critical maintenance (no activities that aren't related to restoring the DG or the offsite source) in the switchyard or on the station's electrical distribution systems.
- Defer maintenance or testing that may impact the reliability of DGs and associated support equipment.
- Implement 10 CFR 50.65(a)(4) fire-specific RMAs associated with the offsite source and the inoperable DG such as maintaining detection, suppression, and fire zone barriers intact and minimizing transient combustibles for those fire areas identified as being significant for the configuration.
- Implement the protected equipment program for the inoperable offsite source and the inoperable DG. This may include limiting access to the switchyard (e.g., locked gates), barricading and/or posting transformer areas, and posting rooms containing intermediate voltage electrical buses, etc.
- Curtail non-essential electrical switching operations to prevent losing the remaining offsite source and to prevent challenging the availability of onsite electrical power.

4.4 Safety-Related 125V DC Battery Inoperable

For TS 3.8.4.B, one DC electrical power source inoperable, RMAs may include:

1) Actions to increase risk awareness and control.

- Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established. Specific focus areas would be to review operating procedures associated with loss of DC power and SBO events.
- Brief the on-shift operations crew concerning the impact that the out-of-service battery would have on the potential response to plant events (e.g., reduced control systems).
- Minimize the accumulation of transient combustibles in accordance with the station fire protection program for the impacted fire zones.
- Minimize activities that could trip the unit (e.g., limit maintenance or testing on reactor protection system instrumentation).
- Challenge any plans to conduct hot work inside the power block or in the switchyard and particularly in the fire zones that have become more important due to the configuration. There should be a bias for not performing hot work during the extended CT (unless it’s directly related to the inoperable battery).

2) Actions to reduce the duration of maintenance activities.

- Confirm parts availability prior to entry into a preplanned RICT.
- Walkdown systems prior to beginning the work.
- Anticipate and be prepared to restore the system as the maintenance concludes.
- Ensure the correct post-maintenance testing and other restoration activities are planned and are ready to be implemented following the maintenance (e.g., to equalize battery voltage with the bus).

3) Actions to minimize the magnitude of the risk increase.

- Proactively implement RMAs during times of high grid stress conditions, such as during high demand conditions.
- Evaluate weather conditions for threats to the reliability of offsite power supplies.
- Defer elective maintenance in the switchyard and on the station’s electrical distribution systems.
- Defer maintenance or testing that affects the reliability of the DGs and their associated support equipment.

- Protect DC electrical buses, remaining operable DC batteries, and other support equipment.

- Implement 10 CFR 50.65(a)(4) fire-specific RMAs for the associated bus and battery.

- Maintain detection, suppression, and fire zone barriers intact and minimize transient combustibles for those fire zones identified as being risk significant for the configuration.

4.5 Electrical Power Distribution Subsystems Inoperable

For TS 3.8.9.A, one or more AC electrical power distribution subsystems inoperable, RMAs may include:

1) Actions to increase risk awareness and control.

- Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established. Specific focus areas may be to review operating procedures and the electrical load list for the inoperable AC distribution subsystem.

- Brief the on-shift operations crew concerning the impact the AC power subsystem has on the potential response to plant events.

- Minimize the accumulation of transient combustibles in accordance with the station fire protection program for the impacted fire zones.

- Challenge hot work in important fire zones. There should be a bias for not performing hot work during the extended CT (unless it's directly related to the inoperable AC electrical power subsystem).

2) Actions to reduce the duration of maintenance activities.

- For preplanned RICT entry, use an equipment outage schedule that identifies and plans all needed resources and provides logic ties between critical activities.

- Confirm parts availability prior to entry into a preplanned RICT.

- Walkdown systems prior to beginning the work.

3) Actions to minimize the magnitude of the risk increase.

- Defer elective maintenance in the switchyard and on the station's electrical distribution systems.

- Defer maintenance or testing that affects the reliability of the DGs or their associated support equipment.

- Protect redundant AC electrical power subsystems and their support equipment.

- Implement 10 CFR 50.65(a)(4) fire-specific RMAs for the associated AC bus.
- Maintain detection, suppression, and fire zone barriers intact and minimize transient combustibles for those fire zones identified as being risk significant for the configuration.

4.6 Low Pressure Injection Subsystem Inoperable

For TS 3.5.2.A, one low pressure injection (LPI) subsystem inoperable, RMAs may include:

1) Actions to increase risk awareness and control.

- Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established. Specific focus areas would be to review procedures associated with reactor coolant system (RCS) leakage or loss of coolant accidents.
- Perform a walkdown and validation of the remaining ECCS systems to validate the standby/readiness condition.
- Minimize the accumulation of transient combustibles in accordance with the station fire protection program for the impacted fire zones.
- Assess the RCS operational leakage rate prior to entering the RICT; consider deferral of a planned RICT if RCS leakage is elevated and/or the source of leakage is unknown.

2) Actions to reduce the duration of maintenance activities.

- For preplanned RICT entry, use an equipment outage schedule that identifies and plans all needed resources and provides logic ties between critical activities.
- Confirm parts availability prior to entry into a preplanned RICT.
- Walkdown systems prior to beginning the work.
- Anticipate and be prepared to restore the system as the maintenance concludes (be prepared to fill and vent systems, rack breakers, etc.)
- Ensure the correct post-maintenance testing is planned and is ready to be implemented following the maintenance (operators are briefed, procedures are ready, etc.)

3) Actions to minimize the magnitude of the risk increase.

- Defer maintenance or testing activities on the redundant ECCS systems and associated support equipment and protect those systems (e.g., post doors to limit access).

- Defer maintenance or testing that affects the reliability of those safety systems that provide defense-in-depth. If testing or maintenance activities must be performed, a review of the potential risk impact should be performed.
- Minimize activities that could trip the unit or increase the frequency of an initiating event.
- Verify system alignment of remaining ECCS systems.
- Implement equipment protection for redundant components and diverse systems.

5.0 References

1. Nuclear Energy Institute (NEI) Topical Report NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, dated October 2012 (ADAMS Accession No. ML12286A322)
2. NRC NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," Revision 1, dated March 2017 (ADAMS Accession No. ML17062A466)
3. EPRI Technical Update 1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," dated December 2012