Exelon Generation

200 Exelon Way
Kennett Square, PA 19348
www.exeloncorp.com

10 CFR 50.90

NMP2L2706

October 31, 2019

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

Nine Mile Point Nuclear Station, Unit 2
Renewed Facility Operating License No. NPF-69
NRC Docket No. 50-410

Subject: 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."

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon) is requesting approval for proposed changes to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. NPF-69 for Nine Mile Point Nuclear Station, Unit 2 (NMP2).

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 was provided by the NRC to the TSTF on November 21, 2018 (ADAMS Accession No. ML18253A085).

- Attachment 1 provides a description and assessment of the proposed changes, 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 the 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 improved Standard Technical Specifications included in TSTF-505, Rev. 2 and the NMP2 plant-specific TS.
- Attachment 5 provides information supporting the redundancy and diversity of instrumentation governed by the TS proposed to be included as part of the Risk Informed Completion Time (RICT) Program in this submittal.
- Attachment 6 provides a list of PRA implementation items that must be completed prior to implementing the RICT Program at NMP2.
- Attachment 7 provides proposed License Condition for NMP2 that require completion of the items listed in Attachment 6 prior to implementation of the RICT program.

These proposed changes have been reviewed and approved by the site's Plant Operations Review Committee in accordance with the requirements of the Exelon Quality Assurance Program.

Exelon requests approval of the proposed amendment by October 31, 2020. The amendment shall be implemented within 180 days following NRC approval, or following completion of the License Condition specified in Attachment 6, on a per unit basis, whichever is later.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (a)(1), 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, "Notice for public comment; State consultation," paragraph (b), Exelon is notifying the State of New York of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning this submittal, please contact Ron Reynolds at (610) 765-5247.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 31st day of October 2019.

Respectfully,

David T. Gudger
Acting Director - Licensing
Exelon Generation Company, LLC

Attachments:

1. Description and Assessment
2. Proposed Technical Specification Changes (Mark-Ups)
3. Proposed Technical Specification Bases Changes (Mark-Ups) (For Information Only)
4. Cross-Reference of TSTF-505 and Nine Mile Point, Unit 2, Technical Specifications
5. Information Supporting Instrumentation Redundancy and Diversity
6. RICT Program PRA Implementation Items
7. Proposed Renewed Facility Operating License Changes (Mark-ups)

Enclosures:

1. List of Revised Required Actions to Corresponding PRA Functions
2. Information Supporting Consistency with Regulatory Guide 1.200, Revision 2
3. Information Supporting Technical Adequacy of PRA Models Without PRA Standards Endorsed by Regulatory Guide 1.200, Revision 2
4. Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models
5. Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)
6. Justification of Application of At-Power PRA Models to Shutdown Modes
7. PRA Model Update Process
8. Attributes of the Real-Time Risk Model
9. Key Assumptions and Sources of Uncertainty
10. Program Implementation
11. Monitoring Program
12. Risk Management Action Examples

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

ATTACHMENT 1

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

<u>Description and Assessment</u>

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon) is requesting approval for proposed changes to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. NPF-69 for Nine Mile Point Nuclear Station, Unit 2 (NMP2).

The proposed amendment would modify the TS requirements related to Completion Times (CTs) for Required Actions (Action allowed outage times for NMP2) to provide the option to calculate a longer, risk-informed CT. A new program, the Risk-Informed Completion Time (RICT) Program, is added to TS Section 5.5, Programs and Manuals.

The methodology for using the RICT Program is described in NEI 06-09, Revision 0-A (hereafter referred to as NEI 06-09-A), "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, which was approved by the NRC on May 17, 2007. 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." However, only those Required Actions described in Attachment 4 and Enclosure 1, as reflected in the proposed TS markups provided in Attachment 2, are proposed to be changed. Some of the modified Required Actions in TSTF-505 are not applicable to NMP2. Also, there are some plant-specific Required Actions not included in TSTF-505 that are included in this proposed amendment.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Exelon has reviewed TSTF-505, Revision 2, and the model safety evaluation dated November 21, 2018 (ADAMS Accession No. ML18253A085). This review included the information provided to support TSTF-505 and the safety evaluation for NEI 06-09-A. As described in the subsequent paragraphs, Exelon has concluded that the technical basis is applicable to NMP2, and support incorporation of this amendment in the NMP2 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 required by Regulatory Guide (RG) 1.200, 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 Nuclear Regulatory Commission.

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 the licensee's programs and procedures that 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 for use in the Real-Time Risk (RTR) tool to assess real-time configuration risk, and describes the scope of, and quality controls applied to, the RTR tool.

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, Section 2.3.2, Step 7.

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

2.3 Optional Changes and Variations

Exelon 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.

NMP2 is a BWR/5 design. The TSTF-505 markups applicable to NMP2 are based on a combination of NUREG-1433, "Standard Technical Specifications General Electric BWR/4 Plants," and NUREG-1434, "Standard Technical Specifications General Electric BWR/6 Plants."

In several instances, the NMP2 TS use different numbering and titles than the Standard Technical Specifications (STS) on which TSTF-505 was based. These differences are administrative and do not affect the applicability of TSTF-505 to the NMP2 TS. Only TS changes consistent with the NMP2 design and TS are included.

Attachment 4 is a cross reference that provides a comparison between the NUREG-1433 or NUREG-1434 Required Actions included in TSTF-505 and the NMP2 Actions included in this license amendment request. The attachment includes a summary description of the referenced Required Actions, which is provided for information purposes only and is not intended to be a verbatim description of the Required Action. The cross reference in Attachment 4 identifies the following:

1. NMP2 Actions that have identical numbers to the corresponding NUREG-1433/NUREG-1434 Required Actions are not deviations from TSTF-505, except for administrative deviations (if any) such as formatting. These deviations are administrative with no impact on the NRC's model safety evaluation dated November 21, 2018.

2. NMP2 Actions that have different numbering than the NUREG-1433/NUREG-1434 Required Actions are an administrative deviation from TSTF-505 with no impact on the NRC's model safety evaluation dated November 21, 2018.

3. For NUREG-1433/NUREG-1434 Required Actions that are not contained in the NMP2 TS, the corresponding TSTF-505 mark-ups for the Required Actions are not applicable to NMP2. This is an administrative deviation from TSTF-505 with no impact on the NRC's model safety evaluation dated November 21, 2018.

4. The model application provided in TSTF-505, Revision 2, includes an attachment for typed, camera-ready (revised) TS pages reflecting the proposed changes. NMP2 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's model safety evaluation dated November 21, 2018. Because of this deviation, the contents and numbering of the attachments for this amendment request differ from the attachments specified in the model application in TSTF-505.

5. There are several plant-specific or BWR/5 design specific LCOs and associated Actions for which NMP2 is proposing to apply the RICT Program that are variations from TSTF-505 as identified in Attachment 4. Attachment 4 was created using the BWR/4 standard from NUREG-1433, with exceptions annotated on Attachment 4 and summarized below. Additional details are contained in Attachment 4 for each individual TS Condition and Action statement that is different the NUREG-1433 BWR/4 standard.

TS 3.3.5.1 - Emergency Core Cooling System (ECCS) Instrumentation. The NMP2 Instrumentation is based on the BWR/5 design and is closest to the NUREG-1434 section 3.3.5.1 for the BWR/6 design. This submittal does not change the instrumentation tables.

TS 3.3.6.1 – Primary Containment Isolation Instrumentation. The NMP2 Instrumentation is based on the BWR/5 design and is closest to the NUREG-1434 section 3.3.5.1 for the BWR/6 design. This submittal does not change the instrumentation tables.

TS 3.3.7.2.2 – Mechanical Vacuum Pump Isolation Instrumentation. This LCO contains four channels of Main Steam Line Radiation – High Function for mechanical vacuum pump isolation Condition is not included in TSTF-505 for either BWR/4 or BWR/6 designs. The system consists of two independent trip systems with two channels in each system. The logic is one-out-of-two-taken-twice. Condition A, one or more channels inoperable, is included in this LAR as it has a restore Action. A Note is added to the LCO Completion Time stating, "Not applicable when trip capability is not maintained." Condition B is a loss of function and is excluded. Condition C is a default Condition and is excluded.

Exelon has determined that the application of a RICT for these NMP2 plant-specific LCOs is consistent with TSTF-505, Revision 2, and with the NRC's model safety evaluation dated November 21, 2018. Application of a RICT for these plant-specific LCOs will be controlled under the RICT Program. The RICT Program provides the necessary administrative controls to permit extension of Completion Times and thereby delay reactor shutdown or remedial actions, if risk is assessed and managed within specified limits and programmatic requirements. The specified safety function or performance levels of TS required structures, systems or components (SSCs) are unchanged, and the remedial actions, including the requirement to shut down the reactor, are also unchanged; only the Completion Times are extended by the RICT Program.

Application of a RICT will be evaluated using the methodology and probabilistic risk guidelines contained in NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," which was approved by the NRC on May 17, 2007 (ADAMS Accession No. ML071200238). The NEI 06-09-A, methodology includes a requirement to perform a quantitative assessment of the potential impact of the application of a RICT on risk, to reassess risk due to plant configuration changes, and to implement compensatory measures and risk management actions (RMAs) to maintain the risk below acceptable regulatory risk thresholds. In addition, the NEI 06-09-A, methodology satisfies the five key safety principles specified in Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications," dated August 1998 (ADAMS Accession No. ML003740176), relative to the risk impact due to the application of a RICT.

Therefore, the proposed application of a RICT in the NMP2 plant-specific Actions is consistent with TSTF-505, Revision 2, and with the NRC staff’s model safety evaluation dated November 21, 2018.

Exelon has reviewed these changes and determined that they do not affect the applicability of TSTF-505, Revision 2, to the NMP2 TS.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Determination

Exelon Generation Company, LLC (Exelon) 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.

Nine Mile Point Nuclear Station, Unit 2 (NMP2), requests adoption of an approved change to the standard technical specifications (STS) and plant-specific technical specifications (TS), 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 6, "Administrative Controls," entitled the "Risk-Informed Completion Time Program."

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

1. Do the proposed changes involve a significant increase in the probability or consequences of 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. Do the proposed changes 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 effects 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, Exelon concludes that the proposed changes present no significant hazards consideration under the 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 EVALUATION

The proposed changes would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed changes.

ATTACHMENT 2

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

<u>Proposed Technical Specification Changes (Mark-Ups)</u>

<u>TS Pages</u>

1.3-13	3.6.1.2-4
3.1.7-1	3.6.1.3-1 through -2
3.3.1.1-1	3.6.1.3-7 through -9
3.3.2.2-1	3.6.1.6-1
3.3.4.1-1	3.6.1.7-1
3.3.4.1-2	3.6.2.3-1
3.3.4.2-1	3.6.2.4-1
3.3.5.1-3 through -7	3.7.1-1
3.3.5.3-1	3.7.1-2
3.3.5.3-2	3.7.5-1
3.3.6.1-1	3.8.1-2 through -4
3.3.7.2-1	3.8.4-1
3.3.8.1-1	3.8.7-1
3.5.1-1	3.8.8-1
3.5.1-2	3.8.8-2
3.5.3-1	5.5-13

1.3 Completion Times

EXAMPLES	EXAMPLE 1.3-7 (continued) 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.
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

Insert A

INSERT A

EXAMPLES (continued)

EXAMPLE 1.3-8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program.
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> 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.

INSERT A

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.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	In accordance with the Surveillance Frequency Control Program

(continued)

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

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

OR
-----------NOTE-----------
Not applicable when trip capability is not maintained.

In accordance with the Risk Informed Completion Time Program

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. OR A.2 Place associated trip system in trip.	12 hours 12 hours
B. --------------NOTE-------------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, and 2.e. ------------------------------------ One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours

OR
-----------NOTE-----------
Not applicable when trip capability is not maintained.

In accordance with the Risk Informed Completion Time Program

OR
-----------NOTE-----------
Not applicable when trip capability is not maintained.

In accordance with the Risk Informed Completion Time Program

(continued)

3.3 INSTRUMENTATION

3.3.2.2 Feedwater System and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Three channels of feedwater system and main turbine high water level trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater system and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days OR -----------NOTE----------- Not applicable when trip capability is not maintained. ---------------------------- In accordance with the Risk Informed Completion Time Program
B. Two or more feedwater system and main turbine high water level trip channels inoperable.	B.1 Restore feedwater system and main turbine high water level trip capability.	2 hours

3.3 INSTRUMENTATION

3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV) – Closure; and
2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure – Low.

OR

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER ≥ 26% RTP with any recirculation pump in fast speed.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status. OR	72 hours OR -----------NOTE----------- Not applicable when trip capability is not maintained. ------------------------------ In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 ------------NOTE------------ Not applicable if inoperable channel is the result of an inoperable breaker. ------------------------------- Place channel in trip.	72 hours OR -----------NOTE----------- Not applicable when trip capability is not maintained. ------------------------------ In accordance with the Risk Informed Completion Time Program
B. One or more Functions with EOC-RPT trip capability not maintained. AND MCPR limit for inoperable EOC-RPT not made applicable.	B.1 Restore EOC-RPT trip capability. OR B.2 Apply the MCPR limit for inoperable EOC-RPT as specified in the COLR.	2 hours 2 hours
C. Required Action and associated Completion Time not met.	C.1 Remove the associated recirculation pump fast speed breaker from service. OR C.2 Reduce THERMAL POWER to < 26% RTP.	4 hours 4 hours

3.3 INSTRUMENTATION

3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

LCO 3.3.4.2 Two channels per trip system for each ATWS – RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level – Low Low, Level 2; and

b. Reactor Vessel Steam Dome Pressure – High.

APPLICABILITY: MODE 1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more channels inoperable.	A.1	Restore channel to OPERABLE status.	14 days OR -----------NOTE----------- Not applicable when trip capability is not maintained. ------------------------------ In accordance with the Risk Informed Completion Time Program
	OR		
	A.2	------------NOTE------------ Not applicable if inoperable channel is the result of an inoperable breaker. -------------------------------- Place channel in trip.	14 days OR -----------NOTE----------- Not applicable when trip capability is not maintained. ------------------------------ In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.1 Place channel in trip.	12 hours for Functions 1.a, 1.d, 2.a, and 2.d AND 24 hours for Functions other than Functions 1.a, 1.d, 2.a, and 2.d OR -----------NOTE----------- Not applicable when trip capability is not maintained. ------------------------------ In accordance with the Risk Informed Completion Time Program
	OR B.3.2 ------------NOTE------------ Only applicable for Functions 1.a, 1.d, 2.a, and 2.d. -------------------------------- Isolate the affected flow path(s).	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	------------NOTE------------ Only applicable for Functions 1.e, 1.f, 1.g, 1.h, 1.i, 1.j, 2.e, 2.f, 2.g, 2.h, and 2.i. ---------------------------------- Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>		
	C.2	Restore channel to OPERABLE status.	24 hours <u>OR</u> -----------NOTE----------- Not applicable when trip capability is not maintained. ------------------------------ In accordance with the Risk Informed Completion Time Program
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1	------------NOTE------------- Only applicable if HPCS pump suction is not aligned to the suppression pool. -------------------------------- Declare HPCS System inoperable.	1 hour from discovery of loss of HPCS initiation capability
	<u>AND</u>		

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2.1	Place channel in trip.	24 hours
	<u>OR</u>		
	D.2.2	Align the HPCS pump suction to the suppression pool.	24 hours
			<u>OR</u> -----------NOTE----------- Not applicable when trip capability is not maintained. ----------------------------- In accordance with the Risk Informed Completion Time Program
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1	------------NOTE------------ Only applicable for Functions 1.k, 1.l, and 2.j. ---------------------------------- Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>		
	E.2	Restore channel to OPERABLE status.	7 days
			<u>OR</u> -----------NOTE----------- Not applicable when trip capability is not maintained. ----------------------------- In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	F.1	Declare Automatic Depressurization System (ADS) valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems or in accordance with the Risk Informed Completion Time Program
	AND		
	F.2	Place channel in trip.	96 hours from discovery of inoperable channel concurrent with HPCS or reactor core isolation cooling (RCIC) inoperable AND 8 days OR --------NOTE--------- Not applicable when trip capability is not maintained. ------------------------- In accordance with the Risk Informed Completion Time Program
G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1	------------NOTE------------ Only applicable for Functions 4.b, 4.d, 4.e, 5.b, and 5.d. -------------------------------- Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	AND		
			(continued)

or in accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. (continued)	G.2 Restore channel to OPERABLE status.	96 hours from discovery of inoperable channel concurrent with HPCS or RCIC inoperable AND 8 days
H. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	H.1 Declare associated supported feature(s) inoperable.	Immediately

OR
--------NOTE--------
Not applicable when trip capability is not maintained.

In accordance with the Risk Informed Completion Time Program

3.3 INSTRUMENTATION

3.3.5.3 Reactor Core Isolation Cooling (RCIC) System Instrumentation

LCO 3.3.5.3

The RCIC System instrumentation for each Function in Table 3.3.5.3-1 shall be OPERABLE.

APPLICABILITY:

MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

--NOTES--

1. Separate Condition entry is allowed for each channel.

2. When the Function 2 channels are placed in an inoperable status solely for performance of SR 3.5.3.4, entry into associated Conditions and Required Actions is not required.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.3-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	B.1 Declare RCIC System inoperable.	1 hour from discovery of loss of RCIC initiation capability
	<u>AND</u>	
<u>OR</u> -----------NOTE----------- Not applicable when trip capability is not maintained. ------------------------------ In accordance with the Risk Informed Completion Time Program	B.2 Place channel in trip.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	C.1 Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	D.1 ------------NOTE------------ Only applicable if RCIC pump suction is not aligned to the suppression pool. ------------------------------- Declare RCIC System inoperable.	1 hour from discovery of loss of RCIC initiation capability
	<u>AND</u>	
	D.2.1 Place channel in trip.	24 hours <u>OR</u> -----------NOTE----------- Not applicable when trip capability is not maintained. ----------------------------- In accordance with the Risk Informed Completion Time Program
	<u>OR</u>	
	D.2.2 Align RCIC pump suction to the suppression pool.	24 hours
E. Required Action and associated Completion Time of Condition B, C, or D not met.	E.1 Declare RCIC System inoperable.	Immediately

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.b, 5.b, and 5.c OR -----------NOTE----------- Not applicable when trip capability is not maintained. ----------------------------- In accordance with the Risk Informed Completion Time Program AND 24 hours for Functions other than Functions 2.b, 5.b, and 5.c OR -----------NOTE----------- Not applicable when trip capability is not maintained. ----------------------------- In accordance with the Risk Informed Completion Time Program
B. One or more automatic Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately

(continued)

3.3 INSTRUMENTATION

3.3.7.2 Mechanical Vacuum Pump Isolation Instrumentation

LCO 3.3.7.2 Four channels of Main Steam Line Radiation – High Function for the mechanical vacuum pump isolation shall be OPERABLE.

APPLICABILITY: MODES 1 and 2 with any mechanical vacuum pump in service and any main steam line not isolated.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status. OR A.2 ------------NOTE------------ Not applicable if inoperable channel is the result of an inoperable isolation valve or mechanical vacuum pump breaker. -------------------------------- Place channel in trip.	12 hours OR --------NOTE-------- Not applicable when trip capability is not maintained. ------------------------- In accordance with the Risk Informed Completion Time Program 12 hours OR -----------NOTE----------- Not applicable when trip capability is not maintained. ------------------------------ In accordance with the Risk Informed Completion Time Program
B. Mechanical vacuum pump isolation capability not maintained.	B.1 Restore isolation capability.	1 hour

(continued)

3.3 INSTRUMENTATION

3.3.8.1 Loss of Power (LOP) Instrumentation

LCO 3.3.8.1 The LOP instrumentation for each Function in Table 3.3.8.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When the associated diesel generator (DG) is required to be OPERABLE by LCO 3.8.2, "AC Sources – Shutdown."

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	1 hour OR -----------NOTE----------- Not applicable when trip capability is not maintained. ------------------------------ In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Declare associated DG inoperable.	Immediately

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS), RPV WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS – Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable to HPCS.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. High Pressure Core Spray (HPCS) System inoperable.	B.1 Verify by administrative means RCIC System is OPERABLE when RCIC is required to be OPERABLE. AND B.2 Restore HPCS System to OPERABLE status.	Immediately 14 days~~*~~ OR In accordance with the Risk Informed Completion Time Program

(continued)

~~* A one-time change to this Completion Time from 14 days to 35 days due to the HPCS DG replacement has been approved via emergency license amendment request. This Completion Time expires on 12/31/2018 at 0100.~~

Delete

Delete

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two ECCS injection subsystems inoperable. OR One ECCS injection and one ECCS spray subsystem inoperable.	C.1 Restore one ECCS injection/spray subsystem to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours
E. One required ADS valve inoperable.	E.1 Restore ADS valve to OPERABLE status.	14 days OR In accordance with the Risk Informed Completion Time Program
F. One required ADS valve inoperable. AND One low pressure ECCS injection/spray subsystem inoperable.	F.1 Restore ADS valve to OPERABLE status. OR F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program 72 hours OR In accordance with the Risk Informed Completion Time Program

(continued)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS), RPV WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.3 RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable to RCIC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCIC System inoperable.	A.1 Verify by administrative means High Pressure Core Spray System is OPERABLE.	Immediately
	<u>AND</u>	
	A.2 Restore RCIC System to OPERABLE status.	14 days <u>OR</u> In accordance with the Risk Informed Completion Time Program.
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	B.2 Reduce reactor steam dome pressure to ≤ 150 psig.	36 hours

3.6 CONTAINMENT SYSTEMS

3.6.1.7 Suppression Chamber-to-Drywell Vacuum Breakers

LCO 3.6.1.7 Each suppression chamber-to-drywell vacuum breaker shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One line with one or more suppression chamber-to-drywell vacuum breakers inoperable for opening.	A.1 Restore the vacuum breaker(s) to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
B. -------------NOTE------------- Separate Condition entry is allowed for each suppression chamber-to-drywell vacuum breaker line. ---------------------------------- One or more lines with one suppression chamber-to-drywell vacuum breaker not closed.	B.1 Close the open vacuum breaker.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more primary containment air locks inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate primary containment overall leakage rate per LCO 3.6.1.1, using current air lock test results.	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 OR --------NOTE--------- Not applicable if leakage exceeds limits or if loss of function. ------------------------- In accordance with the Risk Informed Completion Time Program.
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	12 hours
	AND	
	D.2 Be in MODE 4.	36 hours

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV and each Secondary Containment Bypass Leakage Valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

ACTIONS

--NOTES--

1. 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 systems made inoperable by PCIVs.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------NOTE-------------- Only applicable to penetration flow paths with two or more PCIVs. ------------------------------------ One or more penetration flow paths with one PCIV inoperable except due to leakage not within limit.	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. AND	4 hours except for main steam line OR In accordance with the Risk Informed Completion Time Program AND 8 hours for main steam line OR In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS

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. ----------------------------------	following isolation
		Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside primary containment <u>AND</u> Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.3	Perform SR 3.6.1.3.6 for the resilient seal purge supply valves closed to comply with Required Action D.1.	Once per 92 days
E. One or more penetration flow paths with one or more containment purge exhaust valves not within purge valve leakage limits.	E.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.	24 hours <u>OR</u> --------NOTE--------- Not applicable if there is a loss of function. ------------------------- In accordance with the Risk Informed Completion Time Program.
	<u>AND</u>		

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.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. -------------------------------	following isolation
	Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside containment
	AND	(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued)	E.3	Perform SR 3.6.1.3.6 for the resilient seal purge exhaust valves closed to comply with Required Action E.1.	Once per 92 days following isolation
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met in MODE 1, 2, or 3.	F.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	F.2	Be in MODE 4.	36 hours
G. Required Action and associated Completion Time of Condition A, B, C, D, or E not met for PCIV(s) required to be OPERABLE during MODE 4 or 5.	G.1 .	Initiate action to restore valve(s) to OPERABLE status.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.1.6 Residual Heat Removal (RHR) Drywell Spray System

LCO 3.6.1.6 Two RHR drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1 Restore RHR drywell spray subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Two RHR drywell spray subsystems inoperable.	B.1 Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. Two RHR suppression pool cooling subsystems inoperable.	B.1 Restore one RHR suppression pool cooling subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray

LCO 3.6.2.4 Two RHR suppression pool spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR suppression pool spray subsystem inoperable.	A.1	Restore RHR suppression pool spray subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Two RHR suppression pool spray subsystems inoperable.	B.1	Restore one RHR suppression pool spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 AND C.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

3.7 PLANT SYSTEMS

3.7.1 Service Water (SW) System and Ultimate Heat Sink (UHS)

LCO 3.7.1

a. Division 1 and 2 SW subsystems and UHS shall be OPERABLE.

AND

b.1 Four OPERABLE SW pumps shall be in operation when water temperature of one or two SW subsystem supply headers is ≤ 82°F.

OR

b.2 Five OPERABLE SW pumps shall be in operation when water temperature of one or two SW subsystem supply headers is > 82°F and ≤ 84°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SW supply header cross connect valve inoperable.	A.1 Open the SW supply header cross connect valve.	1 hour
	AND	
	A.2 Restore the SW supply header cross connect valve to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
B. One or more non-safety related SW flow paths with one SW isolation valve inoperable.	B.1 Isolate the affected non-safety related SW flow path(s).	72 hours
C. One SW subsystem inoperable for reasons other than Conditions A and B.	C.1 Restore SW subsystem to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One division of intake deicer heaters inoperable.	D.1 Restore intake deicer heater division to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
E. One required SW pump not in operation.	E.1 Restore required SW pump to operation.	72 hours OR In accordance with the Risk Informed Completion Time Program
F. Two or more required SW pumps not in operation.	F.1 Restore all but one required SW pump to operation.	1 hour OR ---------NOTE------- Not applicable when loss of function can occur. -------------------------- In accordance with the Risk Informed Completion Time Program
G. Required Action and associated Completion Time of Condition A, B, C, D, E, or F not met. OR Both SW subsystems inoperable for reasons other than Conditions A, B, and C. OR UHS inoperable for reasons other than Condition D.	------------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System – Hot Shutdown," for RHR Shutdown Cooling subsystem(s) made inoperable by SW System or UHS. -- G.1 Be in MODE 3. AND G.2 Be in MODE 4.	 12 hours 36 hours

3.7 PLANT SYSTEMS

3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

OR

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Perform a system functional test.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable. AND A.3 Restore required offsite circuit to OPERABLE status.	24 hours from discovery of no offsite power to one division concurrent with inoperability of redundant required feature(s) 72 hours OR In accordance with the Risk Informed Completion Time Program AND 24 hours from discovery of both HPCS and Low Pressure Core Spray (LPCS) Systems with no offsite power OR In accordance with the Risk Informed Completion Time Program
B. One required DG inoperable.	B.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s). AND	1 hour AND Once per 8 hours thereafter (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours
	AND		
	B.4	Restore required DG to OPERABLE status.	72 hours ~~from discovery of an inoperable Division 3 DG~~ ~~AND~~ ~~14 days~~ OR In accordance with the Risk Informed Completion Time Program

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two required offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND	
	C.2 Restore one required offsite circuit to OPERABLE status.	24 hours OR In accordance with the Risk Informed Completion Time Program
D. One required offsite circuit inoperable. AND One required DG inoperable.	------------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.8, "Distribution Systems – Operating," when Condition D is entered with no AC power source to any division. --	
	D.1 Restore required offsite circuit to OPERABLE status.	12 hours OR In accordance with the Risk Informed Completion Time Program
	OR	
	D.2 Restore required DG to OPERABLE status.	12 hours OR In accordance with the Risk Informed Completion Time Program

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources – Operating

LCO 3.8.4 The Division 1, Division 2, and Division 3 DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Division 1 or 2 DC electrical power subsystem inoperable.	A.1 Restore Division 1 and 2 DC electrical power subsystems to OPERABLE status.	2 hours OR In accordance with the Risk Informed Completion Time Program
B. Division 3 DC electrical power subsystem inoperable.	B.1 Declare High Pressure Core Spray System inoperable.	Immediately
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters – Operating

LCO 3.8.7 The Division 1 and Division 2 emergency uninterruptible power supply (UPS) inverters shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One emergency UPS inverter inoperable.	A.1	-----------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.8.8, “Distribution Systems – Operating” with any 120 VAC uninterruptible panel de-energized. --------------------------------	
		Restore emergency UPS inverters to OPERABLE status.	24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

3.8 ELECTRICAL POWER SYTEMS

3.8.8 Distribution Systems – Operating

LCO 3.8.8 The following AC and DC electrical power distribution subsystems shall be OPERABLE:

a. Division 1 and Division 2 AC electrical power distribution subsystems;

b. Division 1 and Division 2 120 VAC uninterruptible electrical power distribution subsystems;

c. Division 1 and Division 2 DC electrical power distribution subsystems; and

d. Division 3 AC and DC electrical power distribution subsystems.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both Division 1 and 2 AC electrical power distribution subsystems inoperable.	A.1 Restore Division 1 and 2 AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>OR</u> --------NOTE--------- Not applicable when loss of function can occur. -------------------------- In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or both Division 1 and 2 120 VAC uninterruptible electrical power distribution subsystems inoperable.	B.1 Restore Division 1 and 2 120 VAC uninterruptible electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>OR</u> ---------NOTE-------- Not applicable when loss of function can occur. -------------------------- In accordance with the Risk Informed Completion Time Program
C. One or both Division 1 and 2 DC electrical power distribution subsystems inoperable.	C.1 Restore Division 1 and 2 DC electrical power distribution subsystem(s) to OPERABLE status.	2 hours <u>OR</u> ---------NOTE-------- Not applicable when loss of function can occur. ------------------------- In accordance with the Risk Informed Completion Time Program
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	12 hours 36 hours
E. One or both Division 3 AC and DC electrical power distribution subsystems inoperable.	E.1 Declare High Pressure Core Spray System inoperable.	Immediately

(continued)

5.5 Programs and Manuals

5.5.13 Control Room Envelope Habitability Program (continued)

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

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.14 Surveillance Frequency Control Program

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

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

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

c. The provision of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

Insert B

INSERT B

5.5.15 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 MODE 1, 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 function(s) performed by the inoperable SSCs.

e. A RICT must be calculated using the following PRA and non-PRA approaches approved by the NRC: full power internal events PRA, internal flooding PRA, fire PRA and seismic bounding value/penalty factor. Changes to these PRA and non-PRA approaches require prior NRC approval. The PRA maintenance and upgrade process will validate that other changes to the PRA models used in the RICT program, including changes involving newly-developed methods, follow ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and NEI 17-07, Revision 2, "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard."

INSERT B

f. A report shall be submitted in accordance with Specification 5.6.9 following each PRA upgrade involving a newly-developed PRA method that has not been previously reported to the NRC for a RICT program and the associated peer review.

5.6 Reporting Requirements (continued)

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and system leakage and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   1. Limiting Condition for Operation 3.4.11, "RCS Pressure and Temperature (P/T) Limits."

   2. Surveillance Requirements 3.4.11.1 through 3.4.11.9

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   1. NEDC-33178P-A, Revision 1, "General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," dated June 2009. The licensee will calculate the fluence for determining the adjusted reference temperature using either; (1) values determined using an NRC-approved, RG 1.190-adherent method, or (2) a fluence estimate, which the licensee has verified as conservative, using an NRC-approved, RG 1.190-adherent method.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.8 OPRM Report

When a report is required by Required Action F.3 of TS 3.3.1.1, “RPS Instrumentation,” a report shall be submitted within the following 90 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to operable status.

Insert C

INSERT C

5.6.9 Probabilistic Risk Assessment (PRA) Upgrade Report

A report shall be submitted following each PRA upgrade involving a newly-developed PRA method that has not been previously reported to the NRC for a RICT program and the associated peer review in accordance with Specification 5.5.15. The report shall describe the scope of the upgrade, including (1) the PRA models upgraded and newly developed methods used, (2) the peer review and finding closure reports available to the NRC for oversight and inspection activities, (3) the number of, and characterization of, the open findings remaining in the upgraded model, and (4) identification of any RICTs of less than 30 days calculated to change by more than 50% for the zero-maintenance configuration.

ATTACHMENT 3

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

Proposed Technical Specification Bases Changes (Mark-Ups)
(for Information Only)

TS Bases Pages

B 3.1.7-3	B 3.3.7.2-4	B 3.8.1-7
B 3.3.1.1-23	B 3.3.8.1-5	B 3.8.1-10
B 3.3.1.1-24	B 3.5.1-6	B 3.8.1-11
B 3.3.1.2-4	B 3.5.1-7	B 3.8.1-12
B 3.3.2.2-4	B 3.5.1-8	B 3.8.1-13
B 3.3.4.1-6	B 3.5.3-3	B 3.8.1-15
B 3.3.4.2-6	B 3.6.1.2-6	B 3.8.1-16
B 3.3.5.1-28	B 3.6.1.3-5	B 3.8.4-4
B 3.3.5.1-30	B 3.6.1.3-11	B 3.8.7-3
B 3.3.5.1-31	B 3.6.1-6	B 3.8.8-5
B 3.3.5.1-33	B 3.6.1.7-3	
B 3.3.5.1-34	B 3.6.2.3-2	
B 3.3.5.1-36	B 3.6.2.4-2	
B 3.3.5.3-8	B 3.7.1-6	
B 3.3.5.3-10	B 3.7.1-7	
B 3.3.6.1-29	B 3.7.1-8	
B 3.3.7.2-3 (for continuity)	B 3.7.5-2	

BASES

APPLICABILITY (continued)

ensure the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to perform its ATWS function during MODES 3, 4, or 5.

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that the radiological consequences of a LOCA involving significant fuel damage remain within the limits of 10 CFR 50.67 (Ref. 5). The SLC System is used to maintain the bulk suppression pool pH at or above 7.0 following a LOCA to ensure that iodine will be retained in the suppression pool water (Ref. 4), as assumed in the Alternative Source Term analyses.

ACTIONS

A.1

or in accordance with the Risk Informed Completion Time Program

If one SLC System subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to shutdown the unit. However, the overall capability is reduced since the remaining OPERABLE subsystem cannot meet the requirements of Reference 1. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of shutting down the unit and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive System to shut down the plant.

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable, given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

(continued)

ACTIONS

A Note has been provided to modify the ACTIONS related to RPS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RPS instrumentation channels provide appropriate compensatory measures for separate, inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel.

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

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Refs. 10 and 11) to permit restoration of any inoperable required channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system (except for Functions 2.a, 2.b, 2.c, 2.d, and 2.e) and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases.) If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a scram or recirculation pump trip (RPT)), Condition D must be entered and its Required Action taken.

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip

(continued)

ACTIONS

B.1 and B.2 (continued)

system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS scram logic for any Function would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in References 10 and 11 for the 12 hour Completion Time, with the exception of Functions 2.a, 2.b, 2.c, 2.d, and 2.e, as described below. Within the 6 hour allowance, the associated Function will have all required channels either OPERABLE or in trip (or in any combination) in one trip system.

Completing one of these Required Actions restores RPS to an equivalent reliability level as that evaluated in References 10 and 11, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels, if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision as to which trip system is in the more degraded state should be based on prudent judgment and current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or RPT, it is permissible to place the other trip system or its inoperable channels in trip.

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

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

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip

(continued)

BASES

ACTIONS

A.1 (continued)

within the Completion Time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in a feedwater pump or main turbine trip), Condition C must be entered and its Required Action taken.

The Completion Time of 7 days is based on the low probability of the event occurring coincident with a single failure in a remaining OPERABLE channel.

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

B.1

With two or more channels inoperable, the Feedwater System and Main Turbine High Water Level Trip Instrumentation cannot perform its design function (feedwater system and main turbine high water level trip capability is not maintained). Therefore, continued operation is only permitted for a 2 hour period, during which feedwater system and main turbine high water level trip capability must be restored. The trip capability is considered maintained when sufficient channels are OPERABLE or in trip such that the feedwater system and main turbine high water level trip logic will generate a trip signal on a valid signal. This requires two channels to each be OPERABLE or in trip. If the required channels cannot be restored to OPERABLE status or placed in trip, Condition C must be entered and its Required Action taken.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of Feedwater System and Main Turbine High Water Level Trip Instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

(continued)

ACTIONS (continued)

inoperable EOC-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable EOC-RPT instrumentation channel.

A.1 and A.2

With one or more required channels inoperable, but with EOC-RPT trip capability maintained (refer to Required Action B.1 and B.2 Bases), the EOC-RPT System is capable of performing the intended function. However, the reliability and redundancy of the EOC-RPT instrumentation is reduced such that a single failure in the remaining trip system could result in the inability of the EOC-RPT System to perform the intended function. Therefore, only a limited time is allowed to restore compliance with the LCO. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of an EOC-RPT, 72 hours is allowed to restore the inoperable channels (Required Action A.1) or apply the EOC-RPT inoperable MCPR limit. Alternately, the inoperable channels may be placed in trip (Required Action A.2) since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted in Required Action A.2, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an RPT), or if the inoperable channel is the result of an inoperable breaker, Condition C must be entered and its Required Actions taken.

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the

(continued)

ACTIONS

A.1 and A.2 (continued)

Functions, and the low probability of an event requiring the initiation of ATWS-RPT, 14 days is provided to restore the inoperable channel (Required Action A.1). Alternately, the inoperable channel may be placed in trip (Required Action A.2), since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. If it is not desirable to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an RPT), or if the inoperable channel is the result of an inoperable breaker, Condition D must be entered and its Required Actions taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining ATWS-RPT trip capability. A Function is considered to be maintaining ATWS-RPT trip capability when sufficient channels are OPERABLE or in trip such that the ATWS-RPT System will generate a trip signal from the given Function on a valid signal, and both recirculation pumps can be tripped. This requires two channels of the Function in the same trip system to each be OPERABLE or in trip, and the corresponding breakers (one fast speed and one LFMG per pump) to be OPERABLE or in trip.

The 72 hour Completion Time is sufficient for the operator to take corrective action (e.g., restoration or tripping of channels) and takes into account the likelihood of an event requiring actuation of the ATWS-RPT instrumentation during this period and the fact that one Function is still maintaining ATWS-RPT trip capability.

(continued)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 12 hours or 24 hours, depending on the Function (12 hours for those Functions that have channel components common to RPS instrumentation and 24 hours for those Functions that do not have channel components common to RPS instrumentation), has been shown to be acceptable (Refs. 5 and 6) to permit restoration of any inoperable channel to OPERABLE status. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.3.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition H must be entered and its Required Action taken. An alternate Required Action is provided (Required Action B.3.2) for Functions 1.a, 1.d, 2.a, and 2.d. For these Functions, in lieu of tripping the channel or entering Condition H (and taking its Required Actions), the affected RHR flow path(s) can be isolated. Isolating the affected flow path(s) accomplishes the safety function of the inoperable channel and does not render the associated LPCI subsystem inoperable. Therefore, this action is acceptable. The allowed Completion Time for isolating the affected flow path(s) is the same as the Completion Time allowed for these Functions in Required Action B.3.1.

C.1 and C.2

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the same variable result in redundant automatic initiation capability being lost for the feature(s). Required Action C.1 features would be those that are initiated by Functions 1.e, I.f, 1.g, 1.h, 1.i, 1.j, 2.e, 2.f, 2.g, 2.h,

(continued)

ACTIONS

C.1 and C.2 (continued)

The Note states that Required Action C.1 is only applicable for Functions 1.e, 1.f, 1.g, 1.h, 1.i, 1.j, 2.e, 2.f, 2.g, 2.h, and 2.i. The Required Action is not applicable to Functions 1.m, 2.k, and 3.i (which also require entry into this Condition if a channel in these Functions is inoperable), since they are the Manual Initiation Functions and are not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 24 hours (as allowed by Required Action C.2) is allowed. Required Action C.1 is also not applicable to Function 3.c (which also requires entry into this Condition if a channel in this Function is inoperable), since the loss of the Function was considered during the development of Reference 5 and considered acceptable for the 24 hours allowed by Required Action C.2.

The Completion Time 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." For Required Action C.1, the Completion Time only begins upon discovery that the same feature in both Divisions (i.e., any Division 1 ECCS and Division 2 ECCS) cannot be automatically initiated due to inoperable channels within the same variable as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 5) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would either cause the initiation or would not necessarily result in a safe state for the channel in all events.

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

(continued)

BASES

ACTIONS (continued)

D.1, D.2.1, and D.2.2

Required Action D.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic component initiation capability for the HPCS System. Automatic HPCS initiation capability is lost if two Function 3.d channels or two Function 3.f channels are inoperable and untripped or if the one Function 3.e channel is inoperable and untripped. In this situation (loss of automatic suction swap), the 24 hour allowance of Required Actions D.2.1 and D.2.2 is not appropriate and the HPCS System must be declared inoperable within 1 hour after discovery of loss of HPCS initiation capability. As noted, the Required Action is only applicable if the HPCS pump suction is not aligned to the suppression pool, since, if aligned, the Function is already performed.

The Completion Time 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." For Required Action D.1, the Completion Time only begins upon discovery that the HPCS System cannot be automatically aligned to the suppression pool due to two inoperable, untripped channels in the same Function. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 5) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2.1 or the suction source must be aligned to the suppression pool per Required Action D.2.2. Placing the inoperable channel in trip performs the intended function of the channel (shifting the suction source to the suppression pool). Performance of either of these two Required Actions will allow operation to continue. If Required Action D.2.1 or Required Action D.2.2 is performed, measures should be taken to ensure that the HPCS System piping remains filled with water. Alternately, if it is not desired to perform Required Actions D.2.1

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

within 24 hours

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

in a loss of the Function (one-out-of-one logic). This loss was considered during the development of Reference 5 and considered acceptable for the 7 days allowed by Required Action E.2.

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

The Completion Time 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." For Required Action E.1, the Completion Time only begins upon discovery that three channels of the variable (Pump Discharge Flow – Low) cannot be automatically initiated due to inoperable channels. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

If the instrumentation that controls the pump minimum flow valve is inoperable such that the valve will not automatically open, extended pump operation with no injection path available could lead to pump overheating and failure. If there were a failure of the instrumentation such that the valve would not automatically close, a portion of the pump flow could be diverted from the reactor injection path, causing insufficient core cooling. These consequences can be averted by the operator's manual control of the valve, which would be adequate to maintain ECCS pump protection and required flow. Furthermore, other ECCS pumps would be sufficient to complete the assumed safety function if no additional single failure were to occur. The 7 day Completion Time of Required Action E.2 to restore the inoperable channel to OPERABLE status is reasonable based on the remaining capability of the associated ECCS subsystems, the redundancy available in the ECCS design, and the low probability of a DBA occurring during the allowed out of service time. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

(continued)

BASES

ACTIONS (continued)

F.1 and F.2

Required Action F.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within similar ADS trip system Functions result in automatic initiation capability being lost for the ADS. Automatic initiation capability is lost if either (a) one or more Function 4.a channels and one or more Function 5.a channels are inoperable and untripped, or (b) one Function 4.c channel and one Function 5.c channel are inoperable and untripped.

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action F.2 is not appropriate, and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability in both trip systems.

The Completion Time 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." For Required Action F.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable, untripped channels within similar ADS trip system Functions as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

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

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. 5) to permit restoration of any inoperable channel to OPERABLE status if both HPCS and RCIC are OPERABLE. If either HPCS or RCIC is inoperable, the time is shortened to 96 hours. If the status of HPCS or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCS or RCIC inoperability. However, total time for an inoperable, untripped channel cannot exceed 8 days. If the status of HPCS or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable, untripped channel. If the inoperable channel cannot be

or in accordance with the Risk Informed Completion Time Program

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

upon discovery that the ADS cannot be automatically initiated due to inoperable channels within similar ADS trip system Functions, as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

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

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. 5) to permit restoration of any inoperable channel to OPERABLE status if both HPCS and RCIC are OPERABLE (Required Action G.2). If either HPCS or RCIC is inoperable, the time is reduced to 96 hours. If the status of HCPS or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCS or RCIC inoperability. However, total time for an inoperable channel cannot exceed 8 days. If the status of HPCS or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

or in accordance with the Risk Informed Completion Time Program

H.1

With any Required Action and associated Completion Time not met, the associated feature(s) may be incapable of performing the intended function and the supported feature(s) associated with the inoperable untripped channels must be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each ECCS instrumentation Function are found in the SRs column of Table 3.3.5.1-1.

(continued)

ACTIONS (continued)

B.1 and B.2

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic initiation capability for the RCIC System (i.e., loss of automatic low water level start capability for Function 1 and loss of automatic high water level trip capability for Function 2). In this case, automatic initiation capability is lost if two channels of a Function, in the same trip system, are inoperable and untripped. In this situation (loss of automatic initiation capability), the 24 hour allowance of Required Action B.2 is not appropriate, and the RCIC System must be declared inoperable within 1 hour after discovery of loss of RCIC initiation capability.

The Completion Time 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." For Required Action B.1, the Completion Time only begins upon discovery that the RCIC System cannot be automatically initiated due to two inoperable untripped Reactor Vessel Water Level – Low Low, Level 2 channels in the same trip system or two inoperable, untripped Reactor Vessel Water Level – High, Level 8 channels in the same trip system. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

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

Because of the redundancy of sensors available to provide initiation signals and the fact that the RCIC System is not assumed in any accident or transient analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 2) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition E must be entered and its Required Action taken.

(continued)

ACTIONS

D.1, D.2.1, and D.2.2 (continued)

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

Because of the redundancy of sensors available to provide initiation signals and the fact that the RCIC System is not assumed in any accident or transient analysis, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 2) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2.1, which performs the intended function of the channel (shifting the suction source to the suppression pool). Alternatively, Required Action D.2.2 allows the manual alignment of the RCIC suction to the suppression pool, which also performs the intended function. If Required Action D.2.1 or D.2.2 is performed, measures should be taken to ensure that the RCIC System piping remains filled with water. If it is not desired to perform Required Actions D.2.1 and D.2.2 (e.g., as in the case where shifting the suction source could drain down the RCIC suction piping), Condition E must be entered and its Required Action taken.

within 24 hours

E.1

With any Required Action and associated Completion Time not met, the RCIC System may be incapable of performing the intended function, and the RCIC System must be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

As noted in the beginning of the SRs, the SRs for each RCIC System instrumentation Function are found in the SRs column of Table 3.3.5.3-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Functions 4 and 5; and (b) for up to 6 hours for Functions 1, 2, and 3 provided the associated Function maintains RCIC initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions

(continued)

BASES

ACTIONS (continued)

allows separate Condition entry for each inoperable primary containment isolation instrumentation channel.

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours, depending on the Function (12 hours for those Functions that have channel components common to RPS instrumentation and 24 hours for those Functions that do not have channel components common to RPS instrumentation), has been shown to be acceptable (Refs. 11 and 12) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Action taken.

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

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic isolation capability being lost for the associated penetration flow path(s). The MSIVs portion of the MSL isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip such that both trip systems will generate a trip signal from the given Function on a valid signal. The MSL drain valves portion of the MSL isolation Functions and the other isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip such that one trip system will generate a trip signal from the given Function on a valid signal. This

(continued)

BASES

APPLICABILITY (continued)

and any main steam line not isolated, to mitigate the consequences of a postulated CRDA. In this condition fission products released during a CRDA could be discharged directly to the environment. Therefore, the mechanical vacuum pump isolation is necessary to assure conformance with the radiological evaluation of the CRDA. In MODE 3, 4 or 5 the consequences of a control rod drop are insignificant, and are not expected to result in any fuel damage or fission product releases. When the mechanical vacuum pump is not in service or the main steam lines are isolated in MODE 1 or 2, fission product releases via this pathway would not occur.

ACTIONS

A Note has been provided to modify the ACTIONS related to Mechanical Vacuum Pump Isolation Instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable Mechanical Vacuum Pump Isolation Instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable Mechanical Vacuum Pump Isolation Instrumentation channel.

A.1 and A.2

With one or more channels inoperable, but with mechanical vacuum pump isolation capability maintained (refer to Required Action B.1 Bases), the Mechanical Vacuum Pump Isolation Instrumentation is capable of performing the intended function. However, the reliability and redundancy of the Mechanical Vacuum Pump Isolation Instrumentation is reduced, such that a single failure in one of the remaining channels could result in the inability of the Mechanical Vacuum Pump Isolation Instrumentation to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to OPERABLE status. Because of the low probability of extensive numbers of inoperabilities affecting multiple channels, and the low probability of an event requiring the initiation of

(continued)

BASES

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

ACTIONS

A.1 and A.2 (continued)

mechanical vacuum pump isolation, 12 hours has been shown to be acceptable (Ref. 5) to permit restoration of any inoperable channel to OPERABLE status (Required Action A.1). Alternately, the inoperable channel, may be placed in trip (Required Action A.2), since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable isolation valve or mechanical vacuum pump breaker, since this may not adequately compensate for the inoperable valve or breaker (e.g., the valve may be inoperable such that it will not close). If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in loss of condenser vacuum), or if the inoperable channel is the result of an inoperable valve or breaker, Condition C must be entered and its Required Actions taken.

B.1

Condition B is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system result in not maintaining mechanical vacuum pump isolation capability. The mechanical vacuum pump isolation capability is maintained when sufficient channels are OPERABLE or in trip such that the Mechanical Vacuum Pump Isolation Instrumentation will generate a trip signal from a valid Main Steam Line Radiation – High signal, and the isolation valve will close and mechanical vacuum pump breakers will open. This would require both trip systems to have one channel OPERABLE or in trip, and the mechanical vacuum pump isolation valve or mechanical vacuum pump breakers to be OPERABLE.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c, 1.d, 1.e, 2.c, 2.d. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) (continued)

per associated emergency bus are available, but only two channels of the Degraded Voltage – 4.16 kV Basis per 4.16 kV emergency bus are required to be OPERABLE when the associated DG is required to be OPERABLE. One channel of each Division 1 and 2 Degraded Voltage – Time Delay, No LOCA Function and Degraded Voltage – Time Delay, LOCA Function per associated emergency bus, is available and required to be OPERABLE when the associated DG is required to be OPERABLE. One channel of the Division 3 Degraded Voltage – Time Delay Function is available and required to be OPERABLE when the associated DG is required to be OPERABLE. These requirements ensure that no single instrument failure can preclude the DG function (Since a failure of a required degraded voltage channel or a time delay channel will only impact the ability of one of the three DGs to start, and only two DGs are credited in the accident analyses, the DG function is still maintained). Refer to LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

ACTIONS

A Note has been provided to modify the ACTIONS related to LOP instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable LOP instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable LOP instrumentation channel.

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

A.1

With one or more required channels of a Function inoperable, the Function may not be capable of performing the intended function. Therefore, only 1 hour is allowed to restore the inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the

(continued)

BASES (continued)

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable HPCS subsystem. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable HPCS subsystem 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.

or in accordance with the Risk Informed Completion Time program

A.1

If any one low pressure ECCS injection/spray subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced because a single failure in one of the remaining OPERABLE subsystems concurrent with a LOCA may result in the ECCS not being able to perform its intended safety function. The 7 day Completion Time is based on a reliability study (Ref. 13) that evaluated the impact on ECCS availability by assuming that various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (i.e., Completion Times).

or in accordance with the Risk Informed Completion Time Program

B.1 and B.2

If the HPCS System is inoperable, and the RCIC System is immediately verified to be OPERABLE (when RCIC is required to be OPERABLE), the HPCS System must be restored to OPERABLE status within 14 days. In this condition, adequate core cooling is ensured by the OPERABILITY of the redundant and diverse low pressure ECCS injection/spray subsystems in conjunction with the ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Immediate verification of RCIC OPERABILITY is therefore required when HPCS is inoperable and RCIC is required to be OPERABLE. This may be performed by an administrative check, by examining logs or other information, to determine if RCIC is out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the RCIC System. However, if the OPERABILITY of the RCIC System cannot be immediately verified and

(continued)

ACTIONS

B.1 and B.2 (continued)

RCIC is required to be OPERABLE, Condition D must be entered. If a single active component fails concurrent with a design basis LOCA, there is a potential, depending on the specific failure, that the minimum required ECCS equipment will not be available. A 14 day Completion Time is based on the results of a reliability study (Ref. 13) and has been found to be acceptable through operating experience.

or in accordance with the Risk Informed Completion Time Program

C.1

With two ECCS injection subsystems inoperable or one ECCS injection and one ECCS spray subsystem inoperable, at least one ECCS injection/spray subsystem must be restored to OPERABLE status within 72 hours. In this condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced in this Condition because a single failure in one of the remaining OPERABLE subsystems concurrent with a design basis LOCA may result in the ECCS not being able to perform its intended safety function. Since the ECCS availability is reduced relative to Condition A, a more restrictive Completion Time is imposed. The 72 hour Completion Time is based on a reliability study, as provided in Reference 13.

D.1 and D.2

If any Required Action and associated Completion Time of Condition A, B, or C are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

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

The LCO requires six ADS valves to be OPERABLE to provide the ADS function. Reference 14 contains the results of an analysis that evaluated the effect of two of seven ADS valves being out of service. This analysis showed that assuming a failure of the HPCS System, operation of only five ADS valves will provide the required depressurization. However, overall reliability of the ADS is reduced because a single failure in the OPERABLE ADS valves could result in a reduction in depressurization capability. Therefore, operation is only allowed for a limited time. The 14 day Completion Time is based on a reliability study (Ref. 13) and has been found to be acceptable through operating experience.

(continued)

BASES

ACTIONS (continued)

F.1 and F.2

If any one low pressure ECCS injection/spray subsystem is inoperable in addition to one required ADS valve inoperable, adequate core cooling is ensured by the OPERABILITY of HPCS and the remaining low pressure ECCS injection/spray subsystems. However, the overall ECCS reliability is reduced because a single active component failure concurrent with a design basis LOCA could result in the minimum required ECCS equipment not being available. Since both a high pressure (ADS) and low pressure subsystem are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the low pressure ECCS injection/spray subsystem or the ADS valve to OPERABLE status. This Completion Time is based on a reliability study (Ref. 13) and has been found to be acceptable through operating experience.

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

G.1 and G.2

If any Required Action and associated Completion Time of Condition E or F are not met or if two or more required ADS valves are inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig 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.

H.1

When multiple ECCS subsystems are inoperable, as stated in Condition H, the plant is in a condition outside of the design basis. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

or in accordance with the Risk Informed Completion Time Program

BASES

ACTIONS

A.1 and A.2

If the RCIC System is inoperable during MODE 1, or MODES 2 or 3 with reactor steam dome pressure > 150 psig, and the HPCS System is immediately verified to be OPERABLE, the RCIC System must be restored to OPERABLE status within 14 days.

In this condition, loss of the RCIC System will not affect the overall plant capability to provide makeup inventory at high RPV pressure since the HPCS System is the only high pressure system assumed to function during a loss of coolant accident (LOCA). OPERABILITY of the HPCS is therefore immediately verified when the RCIC System is inoperable. This may be performed as an administrative check, by examining logs or other information, to determine if the HPCS is out of service for maintenance or other reasons. Verification does not require performing the Surveillances needed to demonstrate the OPERABILITY of the HPCS System. If the OPERABILITY of the HPCS System cannot be immediately verified, however, Condition B must be entered. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCS) is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of RPV water level. Therefore, a limited time is allowed to restore the inoperable RCIC to OPERABLE status.

The 14 day Completion Time is based on a reliability study (Ref. 4) that evaluated the impact on ECCS availability, assuming that various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (AOTs). Because of the similar functions of the HPCS and RCIC, the AOTs (i.e., Completion Times) determined for the HPCS are also applied to RCIC.

B.1 and B.2

If the RCIC System cannot be restored to OPERABLE status within the associated Completion Time, or if the HPCS System is simultaneously inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig 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.

(continued)

ACTIONS

B.1, B.2, and B.3 (continued)

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas or areas with limited access due to inerting and allows these doors to be verified locked closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

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 containment overall leakage rates using current air lock leakage test results. An evaluation is acceptable since it is overly conservative to immediately declare the primary 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) primary containment remains OPERABLE, yet only 1 hour (according to LCO 3.6.1.1) 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 primary containment air locks must be verified closed. This Required Action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1.1, which require that primary containment be restored to OPERABLE status within 1 hour.

Additionally, the air lock must be restored to OPERABLE status within 24 hours (Required Action C.3). The 24 hour Completion Time is reasonable for restoring an inoperable air lock to OPERABLE status considering that at least one door is maintained closed in each affected air lock.

(continued)

or in accordance with the Risk Informed Completion Time Program

or in accordance with the Risk Informed Completion Time Program

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

ACTIONS

A.1 and A.2 (continued)

closed manual valve, a blind flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to the primary containment. The Required Action must be completed within the 4 hour Completion Time (8 hours for main steam lines). The specified time period of 4 hours is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. For main steam lines, an 8 hour Completion Time is allowed. The Completion Time of 8 hours for the main steam lines allows a period of time to restore the MSIVs to OPERABLE status given the fact that MSIV closure will result in isolation of the main steam line(s) and a potential for plant shutdown.

following isolation

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This is necessary to ensure that primary 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 devices outside the primary containment and capable of being mispositioned are in the correct position. The Completion Time for this verification of "once per 31 days for isolation devices outside primary containment" is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. For devices inside the primary containment the specified time period of "prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days," is based on engineering judgment and is considered reasonable in view of the inaccessibility of the devices and the existence of other administrative controls ensuring that device misalignment is an unlikely possibility.

Condition A is modified by a Note indicating that this Condition is only applicable to those penetration flow paths

(continued)

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

ACTIONS

E.1, E.2, and E.3 (continued)

is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.

following isolation

In accordance with Required Action E.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are 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 valve manipulation. Rather, it involves verification that those isolation devices outside containment and potentially capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, 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.

Required Action E.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows them 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 once they have been verified to be in the proper position, is low.

following isolation

For the containment purge exhaust valve with resilient seal that is closed in accordance with Required Action E.1, SR 3.6.1.3.6 must be performed at least once every 92 days. This provides assurance that degradation of the resilient seal is detected and confirms that the leakage rate of the

(continued)

BASES

LCO (continued)

performing their intended design function). The requirement that the vacuum breakers be closed ensures that there is no excessive bypass leakage should a LOCA occur.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell floor, caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture that purges the drywell of air and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3. Excessive negative pressure inside the drywell could occur due to inadvertent actuation of drywell sprays.

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES; therefore, maintaining suppression chamber-to-drywell vacuum breakers OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one line with one or more vacuum breakers inoperable for opening (e.g., a vacuum breaker is not open and may be stuck closed or not within its opening setpoint limit, so that it would not function as designed during an event that depressurized the drywell), the remaining OPERABLE vacuum breakers in the other three lines are capable of providing the vacuum relief function. However, overall system reliability is reduced because a single failure in one of the remaining vacuum breakers in the other three lines could result in an excessive suppression chamber-to-drywell differential pressure during a DBA. Therefore, with one or more of the vacuum breakers inoperable in one line, 72 hours is allowed to restore the inoperable vacuum breaker(s) to OPERABLE status so that plant conditions are consistent with those assumed for the design basis analysis. The 72 hour Completion Time is considered acceptable due to the low probability of an event in which the remaining vacuum breaker capability would not be adequate.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

suppression pool temperature is calculated to remain below the design limit.

The RHR Suppression Pool Cooling System satisfies Criterion 3 of Reference 2.

LCO

During a DBA, a minimum of one RHR suppression pool cooling subsystem is required to maintain the primary containment peak pressure and temperature below the design limits (Ref. 1). To ensure that these requirements are met, two RHR suppression pool cooling subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE, assuming the worst case single active failure. An RHR suppression pool cooling subsystem is OPERABLE when the pump, a heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE. Management of gas voids is important to RHR Suppression Pool Cooling System OPERABILITY.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause both a release of radioactive material to primary containment and a heatup and pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, the RHR Suppression Pool Cooling System is not required to be OPERABLE in MODE 4 or 5.

ACTIONS

A.1

or in accordance with the Risk Informed Completion Time Program

With one RHR suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining RHR suppression pool cooling subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time is acceptable in light of the redundant RHR suppression pool cooling capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	Reference 1 contains the results of analyses used to predict primary containment pressure and temperature following large and small break loss of coolant accidents. The intent of the analyses is to demonstrate that the pressure reduction capacity of the RHR Suppression Pool Spray System (in conjunction with the RHR Drywell Spray System) is adequate to maintain the primary containment conditions within design limits. The time history for primary containment pressure is calculated to demonstrate that the maximum pressure remains below the design limit. The RHR suppression pool spray satisfies Criterion 3 of Reference 2.
LCO	In the event of a DBA, a minimum of one RHR suppression pool spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak pressure below the design limits (Ref. 1). To ensure that these requirements are met, two RHR suppression pool spray subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR suppression pool spray subsystem is OPERABLE when one pump and associated piping, valves, instrumentation, and controls are OPERABLE. Management of gas voids is important to RHR Suppression Pool Spray System OPERABILITY.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining RHR suppression pool spray subsystems OPERABLE is not required in MODE 4 or 5.
ACTIONS	A.1 With one RHR suppression pool spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program. In this condition, the remaining OPERABLE RHR suppression pool spray subsystem is adequate to perform the primary containment bypass leakage mitigation function.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, and 3, the SW System and UHS are required to be OPERABLE to support OPERABILITY of the equipment serviced by the SW System and UHS, and are required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the SW System and UHS are determined by the systems they support and therefore, the requirements are not the same for all facets of operation in MODES 4 and 5. Thus, the LCOs of the systems supported by the SW System and UHS will govern SW System and UHS OPERABILITY requirements in MODES 4 and 5.

ACTIONS

A.1 and A.2

If one SW supply header cross connect valve is inoperable due to being closed, an assumption of the LOCA analysis is not met. Therefore, the SW supply header cross connect valve must be opened within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem and takes into account the low probability of a LOCA occurring during this time. If one SW supply header cross connect valve is inoperable due to being incapable of automatically closing, the remaining OPERABLE SW supply header cross connect valve is adequate to ensure the two Divisions can be split, thus ensuring the SW heat removal function is maintained during a LOOP or LOOP/LOCA. However, the overall reliability is reduced because a single failure of the OPERABLE SW supply header cross connect valve could result in loss of the SW function during a LOOP or LOOP/LOCA. Therefore, the SW supply header cross connect valve must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE SW supply header cross connect valve and the low probability of a LOOP or LOOP/LOCA occurring during this period.

(continued)

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

ACTIONS (continued)

B.1

When one or more non-safety related SW flow paths (i.e., the two supply and two return flow paths) have one SW isolation valve that is inoperable, the remaining OPERABLE SW isolation valve in each affected non-safety related SW flow path is adequate to ensure the non-safety related flow paths can be isolated from the safety related flow paths, thus ensuring the SW heat removal function is maintained. However, the overall reliability is reduced because a single failure of the OPERABLE SW isolation valve in an affected non-safety related flow path could result in loss of the SW function during a LOOP or LOOP/LOCA. Therefore, the affected non-safety related flow path(s) must be isolated within 72 hours. Isolating the affected non-safety related flow path(s) is acceptable since this action performs the function of the SW isolation valves. The 72 hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE SW isolation valve in each affected non-safety related flow path and the low probability of a LOOP or LOOP/LOCA occurring during this period.

C.1

If one SW subsystem is inoperable for reasons other than Conditions A and B (e.g., one or two required SW pumps inoperable in one subsystem), it must be restored to OPERABLE status within 72 hours. With the unit in this condition, the remaining OPERABLE SW subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SW subsystem could result in a loss of the SW function. The 72 hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

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

D.1

If one division of intake deicer heaters is inoperable, it must be restored to OPERABLE status within 72 hours. With the unit in this condition, the remaining OPERABLE deicer heater division is adequate to ensure both intake structures

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

(continued)

ACTIONS

D.1 (continued)

remain free of ice blockage, thus ensuring the UHS is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE intake deicer heater division could result in loss of the UHS function during a DBA. The 72 hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE intake deicer heater division and the low probability of a DBA occurring during this period.

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

E.1

If one required SW pump is not in operation, it must be restored to operation within 72 hours. With the unit in this condition, the remaining operating SW pumps are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure of a remaining operating pump could result in loss of the SW function during a DBA LOCA. The 72 hour Completion Time was developed taking into account the redundant capabilities afforded by the operating pumps and the low probability of a DBA LOCA occurring during this period.

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

F.1

If two or more required SW pumps are not in operation, all but one of the required SW pumps must be in operation within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem and is consistent with the 1 hour provided in LCO 3.0.3. This time period also takes into account the low probability of a DBA LOCA occurring during this time.

G.1 and G.2

If any Required Action and associated Completion Time of Condition A, B, C, D, E, or F are not met, or both SW subsystems are inoperable for reasons other than Conditions A, B, and C, or the UHS is inoperable for reasons other than Condition D, 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 12 hours and in MODE 4 within 36 hours.

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, such that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow continued operation.

An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2). The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR.

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 23% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.2, sufficient margin to this limit exists < 23% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), and the MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program if the inoperability of the Main Steam Turbine Bypass System is not the result of APLHGR or MCPR limit malfunctions.

ACTIONS

A.2 (continued)

Completion Time for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before the unit is subjected to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection may have been lost for the required feature's function; however, function is not lost. 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

According to Regulatory Guide 1.93 (Ref. 8), operation may continue in Condition A for a period that should not exceed 72 hours. Alternatively, a Completion Time can be determined in accordance with the Risk Informed Completion Time Program.

This Completion Time assumes sufficient offsite power remains to power the minimum loads needed to respond to analyzed events. In the event both the HPCS and Low Pressure Core Spray (LPCS) Systems are without offsite power, this assumption is not met. Therefore, the optional Completion Time is specified. Should the HPCS and LPCS Systems be affected, the 24 hour Completion Time is conservative with respect to the Regulatory Guide assumptions supporting a 24 hour Completion Time for both offsite circuits inoperable. 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 plant safety systems. In this condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E distribution system.

(continued)

ACTIONS

B.2 (continued)

required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DGs. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG(s), SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DGs, the other DGs are declared inoperable upon discovery, and Condition E or G of LCO 3.8.1 is entered, as applicable. Once the failure is repaired, and the common cause failure no longer exists, Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG(s), performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of those DG(s).

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the Deficiency Event Report Program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 9), 24 hours is reasonable time to confirm that the OPERABLE DG(s) are not affected by the same problem as the inoperable DG.

B.4

In Condition B, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E distribution system. ~~Although Condition B applies to a single inoperable DG, several Completion Times are specified for this Condition.~~

~~The first Completion Time applies to an inoperable Division 3 DG.~~ The 72 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.

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

(continued)

Delete

BASES

ACTIONS

B.4 (continued)

This Completion Time begins only “upon discovery of an inoperable Division 3 DG” and, as such, provides an exception to the normal “time zero” for beginning the allowed outage time “clock” (i.e., for beginning the clock for an inoperable Division 3 DG when Condition B may have already been entered for another equipment inoperability and is still in effect).

The second Completion Time (14 days) applies to an inoperable Division 1 or Division 2 DG and is a risk-informed Completion Time based on a plant-specific risk analysis. The extended Completion Time would typically be used for voluntary planned maintenance or inspections but can also be used for corrective maintenance. However, use of the extended Completion Time for voluntary planned maintenance should be limited to once within an operating cycle (24 months) for each DG (Division 1 and Division 2). When utilizing an extended DG Completion Time (greater than 72 hours and up to 14 days), the compensatory measures and configuration risk management controls listed below shall be implemented. For planned maintenance utilizing an extended Completion Time, these measures and controls shall be implemented prior to entering Condition B. For an unplanned entry into an extended Completion Time, these measures and controls shall be implemented without delay.

a. The other two DGs are operable and no planned maintenance or testing activities are scheduled on those two DGs.

b. No planned maintenance or testing activities are scheduled in Scriba Substation, the NMP2 115 kV switchyard, or on the 115 kV power supply lines and transformers which could cause a line outage or challenge offsite power availability.

c. The HPCS system is operable and no planned maintenance or testing activities are scheduled.

d. The RCIC system is operable and no planned maintenance or testing activities are scheduled.

e. The NMP2 and Nine Mile Point Unit 1 (NMP1) diesel-driven fire pumps and the cross-tie between the NMP2 and NMP1 fire protection water supply systems are available to provide a backup cooling water supply to the Division 3 DG and no planned maintenance or testing activities are scheduled.

(continued)

Delete

BASES

ACTIONS

B.4 (continued)

f. The Division 1 and Division 2 Residual Heat Removal (RHR) pumps and the LPCS pump are operable and no planned maintenance or testing activities are scheduled.

g. Both divisions of the redundant reactivity control system and the standby liquid control system (equipment required for mitigation of Anticipated Transients Without Scram (ATWS) events) are operable and no planned maintenance or testing activities are scheduled.

h. The stability of existing and projected grid conditions will be confirmed prior to planned entry into the extended DG Completion Time by contacting the Transmission System Operator (TSO).

i. Operating crews will be briefed on the DG work plan. As a minimum, the briefing will include the important procedural actions that could be required in the event a loss of offsite power, station blackout, or fire condition occurs.

j. The extended DG Completion Time will not be entered for planned maintenance if severe weather conditions (high winds, tornado, or heavy snow/ice) with the potential to degrade or limit offsite power availability are present, or if official weather forecasts are predicting such conditions to occur.

k. Except for the room housing the inoperable DG, no hot work permits will be active for the control building or the normal switchgear rooms.

l. A portable generator is available as a temporary backup source of AC power to one of the Division 1 or Division 2 battery chargers and is pre-staged within the protected area near the NMP2 control building.

m. Four portable power supplies are available for use to facilitate operation of safety relief valves to maintain reactor coolant system pressure control for an extended station blackout condition and are verified to be functional.

If one or more of the above measures and controls are not met while in the extended Completion Time, enter the condition into the corrective action program, follow applicable Required Actions, manage the risk in accordance with the configuration risk management program, and initiate actions to restore the measure(s) or control(s) without delay.

(continued)

Delete

BASES

ACTIONS

B.4 (continued)

Similar to Required Action B.2, the Completion Time of Required Action B.4 allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This exception results in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered.

C.1 and C.2

Required Action C.1 addresses actions to be taken in the event of concurrent failure of redundant required features. Required Action C.1 reduces the vulnerability to a loss of function. The Completion Time for taking these actions is reduced to 12 hours from that allowed with only one division without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 8) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety divisions 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. These features are designed with redundant safety related divisions (i.e.,

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Regulatory Guide 1.93 (Ref. 8), with the available offsite AC sources two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

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

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to any division (i.e., the division is de-energized), Actions for LCO 3.8.8, "Distribution Systems – Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of the offsite circuit and one DG without regard to whether a division is de-energized. LCO 3.8.8 provides the appropriate restrictions for a de-energized division.

According to Regulatory Guide 1.93 (Ref. 8), operation may continue in Condition D for a period that should not exceed 12 hours. In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

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 low probability of a DBA occurring during this period.

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

E.1

With two DGs inoperable, there is one remaining standby AC source. 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 the majority of ESF equipment at 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. 8), with two required DGs inoperable, operation may continue for a period that should not exceed 2 hours. This Completion Time assumes complete loss of onsite (DG) AC capability to power the minimum loads needed to respond to analyzed events. In the event Division 3 DG in conjunction with Division 1 or 2 DG is inoperable, with the other Division 1 or 2 DG remaining, a significant spectrum of breaks would be capable of being responded to with onsite power. Even the worst case event would be mitigated to some extent – an extent greater than a typical two division design in which this condition represents complete loss of onsite power function. Given the remaining function, a 24 hour Completion Time is

(continued)

BASES (continued)

ACTIONS

A.1

Condition A represents one division 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 division. The 2 hour limit is consistent with the allowed time for an inoperable DC distribution system division.

If one of the required Division 1 or 2 DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable required battery charger, or inoperable required battery charger and associated inoperable battery), the remaining DC electrical power subsystems have 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 subsystems, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 8) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

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

With the Division 3 DC electrical power subsystem inoperable, the HPCS System may be incapable of performing its intended function and must be immediately declared inoperable. This declaration also requires entry into applicable Conditions and Required Actions of LCO 3.5.1, "ECCS – Operating."

C.1 and C.2

If the DC electrical power subsystem cannot be restored to OPERABLE status 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 plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant

(continued)

BASES

APPLICABILITY (continued)

In MODES 4 and 5, the emergency UPS inverters are not required to be OPERABLE since, during these MODES, if a loss of offsite power occurred (which could result in loss of power to the uninterruptible panels until the DG starts and energizes the associated emergency buses) coincident with an accident requiring the ECCS instrumentation to perform their function, the response time of the ECCS subsystems (which will be delayed due to the loss of power to the uninterruptible panels) is not as critical.

ACTIONS

A.1

With a required emergency UPS inverter inoperable, its associated 120 VAC uninterruptible panels become inoperable until they are re-energized from their Class 1E regulating transformer (maintenance transformer) or emergency UPS inverter using the internal AC source. LCO 3.8.8 addresses this action; however, pursuant to LCO 3.0.6, these actions would not be entered even if the 120 VAC uninterruptible panels were de-energized. Therefore, the ACTIONS are modified by a Note stating that ACTIONS for LCO 3.8.8 must be entered immediately. This ensures the uninterruptible panels are re-energized within 8 hours.

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

Required Action A.1 allows 24 hours to fix the inoperable emergency UPS inverter and return it to service or, alternatively, to place into service the standby emergency UPS inverter for the affected division. 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 plant is exposed because of the inverter inoperability. This risk has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems that such a shutdown might entail. When the 120 VAC uninterruptible panels are powered from their constant voltage maintenance source (or the internal AC source/rectifier with the DC source inoperable), they are relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the 120 VAC uninterruptible panels is the preferred source for powering instrumentation trip setpoint devices.

(continued)

ACTIONS

A.1 (continued)

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 power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, motor control centers, and distribution panels must be restored to OPERABLE status within 8 hours.

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

The Condition A worst scenario is one division without AC power (i.e., no offsite power to the division and the associated DG inoperable). In this situation, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operators' attention be focused on minimizing the potential for loss of power to the remaining division by stabilizing the unit and restoring power to the affected division. 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 operators' attention is diverted from the evaluations and actions necessary to restore power to the affected division to the actions associated with taking the unit to shutdown within this time limit.

b. The low potential for an event in conjunction with a single failure of a redundant component in the division with AC power. (The redundant component is verified OPERABLE in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP).")

(continued)

ATTACHMENT 4

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

Cross-Reference of TSTF-505 and Nine Mile Point Unit 2 Technical Specifications

Cross-Reference of TSTF-505 and Nine Mile Point Nuclear Station, Unit 2 Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	NMP2 Tech Spec	Apply RICT?	Comments
Completion Times	**1.3**	**1.3**		
Example 1.3-8	1.3-8	1.3-8		TSTF-505 changes are incorporated.
Standby Liquid Control (SLC) System	**3.1.7**	**3.1.7**		
One SLC subsystem inoperable.	3.1.7.B	3.1.7.A	Yes	TSTF-505 changes are incorporated.
Reactor Protection System (RPS) Instrumentation	**3.3.1.1**	**3.3.1.1**		
One or more required channels inoperable.	3.3.1.1.A.1 3.3.1.1.A.2	3.3.1.1.A.1 3.3.1.1.A.2	Yes Yes	TSTF-505 changes are incorporated. TSTF-505 changes are incorporated.
One or more Functions with one or more required channels inoperable in both trip systems.	3.3.1.1.B.1 3.3.1.1.B.2	3.3.1.1.B.1 3.3.1.1.B.2	Yes Yes	TSTF-505 changes are incorporated. TSTF-505 changes are incorporated.
Source Range Monitor (SRM) Instrumentation	**3.3.1.2**	**3.3.1.2**		
One or more required SRMs inoperable in MODE 2 with intermediate range monitors (IRMs) on Range 2 or below.	3.3.1.2.A	3.3.1.2.A	No	TSTF-505 changes are excluded. This function is not modeled. The FPIE PRA model considers at-power conditions. The SRMs are used early in startup when the unit is not producing significant power and decay heat is low.

Cross-Reference of TSTF-505 and
Nine Mile Point Nuclear Station, Unit 2 Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	NMP2 Tech Spec	Apply RICT?	Comments
Feedwater and Main Turbine High Water Level Trip Instrumentation	**3.3.2.2**	**3.3.2.2**		
One feedwater and main turbine high water level trip channel inoperable.	3.3.2.2.A	3.3.2.2.A	Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.
Two or more feedwater and main turbine high water level trip channels inoperable.	3.3.2.2.B	3.3.2.2.B.1	No	TSTF-505 changes are excluded due to loss of function.
End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation	**3.3.4.1**	**3.3.4.1**		
One or more required channels inoperable.	3.3.4.1.A.1 3.3.4.1.A.2	3.3.4.1.A.1 3.3.4.1.A.2	Yes Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.
Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation	**3.3.4.2**	**3.3.4.2**		
One or more channels inoperable.	3.3.4.2.A.1 3.3.4.2.A.2	3.3.4.2.A.1 3.3.4.2.A.2	Yes Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.

Cross-Reference of TSTF-505 and
Nine Mile Point Nuclear Station, Unit 2 Technical Specifications

Tech Spec Description	**TSTF-505 Tech Spec**	**NMP2 Tech Spec**	**Apply RICT?**	**Comments**
Emergency Core Cooling System (ECCS) Instrumentation	**3.3.5.1**	**3.3.5.1**		NMP2 aligns with the BWR/6, NUREG-1434, for ECCS instrumentation specifications.
As required by Required Action A.1 and referenced in Table 3.3.5.1-1 (Functions 1.a, 1.b, 2.a, and 2.b; 3.a and 3.b).	3.3.5.1.B.3	3.3.5.1.B.3.1	Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.
As required by Required Action A.1 and referenced in Table 3.3.5.1-1 (Functions 1.c, 2.c, 2.d, and 2.f; 1.e, 2.h, and 3.g).	3.3.5.1.C.2	3.3.5.1.C.2	Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.
As required by Required Action A.1 and referenced in Table 3.3.5.1-1 (Functions 3.d and 3.e).	3.3.5.1.D.2.1	3.3.5.1.D.2.1	Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.
As required by Required Action A.1 and referenced in Table 3.3.5.1-1 (Functions 1.d, 2.g, and 3.f).	3.3.5.1.E.2	3.3.5.1.E.2	Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.
As required by Required Action A.1 and referenced in Table 3.3.5.1-1 (Functions 4.a, 4.b, 4.d, 5.a, 5.b, and 5.d).	3.3.5.1.F.2	3.3.5.1.F.2	Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.

Cross-Reference of TSTF-505 and
Nine Mile Point Nuclear Station, Unit 2 Technical Specifications

Tech Spec Description	**TSTF-505 Tech Spec**	**NMP2 Tech Spec**	**Apply RICT?**	**Comments**
As required by Required Action A.1 and referenced in Table 3.3.5.1-1 (Functions 4.c, 4.e, 4.f, 4.g, 5.c, 5.e, 5.f, and 5.g; 4.h and 5.h).	3.3.5.1.G.2	3.3.5.1.G.2	Yes	TSTF-505 changes are incorporated.
Reactor Core Isolation Cooling (RCIC) System Instrumentation	**3.3.5.2**	**3.3.5.3**		Variation in number sequence due to NMP2 has incorporated RPV WIC as 3.3.5.2 and renumbered RCIC to 3.3.5.3.
As required by Required Action A.1 and referenced in Table 3.3.5.2-1 (Function 1).	3.3.5.2.B.2	3.3.5.3.B.2	Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.
As required by Required Action A.1 and referenced in Table 3.3.5.2-1 (Functions 3 and 4).	3.3.5.2.D.2.1	3.3.5.3.D.2.1	Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.
Primary Containment Isolation Instrumentation	**3.3.6.1**	**3.3.6.1**		NMP2 aligns with the BWR/6, NUREG-1434, for Primary Containment Isolation Instrumentation specifications.
One or more required channels inoperable (Functions 2.a, 2.b, 6.b, 7.a, and 7.b; Functions other than Functions 2.a, 2.b, 6.b, 7.a, and 7.b).	3.3.6.1.A	3.3.6.1.A.1	Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.
Low-Low-Set (LLS) Instrumentation	**3.3.6.3**	**---------**		The NMP2 TS do not contain this TS. Therefore, a change is not proposed to the NMP2 TS.

Cross-Reference of TSTF-505 and
Nine Mile Point Nuclear Station, Unit 2 Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	NMP2 Tech Spec	Apply RICT?	Comments
Mechanical Vacuum Pump Isolation Instrumentation	---------	**3.3.7.2**		
One or more channels inoperable.	--------- ---------	**3.3.7.2.A.1** **3.3.7.2.A.2**	Yes Yes	For Required Actions A.1 and A.2, TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Times which prohibits applying a RICT when trip capability is not maintained.
Loss of Power (LOP) Instrumentation	**3.3.8.1**	**3.3.8.1**		
One or more channels inoperable.	3.3.8.1.A	3.3.8.1.A.1	Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.
Safety/Relief Valves (S/RVs)	**3.4.3**	**3.4.4**		
One [or two] required S/RV(s) inoperable.	3.4.3.A	---------	No	The NMP2 TS do not contain this Required Action as a Restore Action. Therefore, a change is not proposed to the NMP2 TS.
ECCS - Operating	**3.5.1**	**3.5.1**		NMP2 has High Pressure Core Spray (HPCS) System. The notes on this page associated with Amendment 174 are being removed. This change is administrative and acceptable.
One low pressure ECCS injection/spray subsystem inoperable or one LPCI pump in both LPCI subsystems inoperable.	3.5.1.A	3.5.1.A.1	Yes	TSTF-505 changes are incorporated.

Cross-Reference of TSTF-505 and
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Tech Spec Description	TSTF-505 Tech Spec	NMP2 Tech Spec	Apply RICT?	Comments
HPCI System inoperable.	3.5.1.C.2	3.5.1.B.2	Yes	TSTF-505 changes are incorporated. NMP2 BWR/5 design utilizes High Pressure Core Spray (HPCS), not HPCI system.
HPCI System inoperable and Condition A entered.	3.5.1.D.1 3.5.1.D.2	-------- 3.5.1.C.1	No Yes	The NMP2 TS do not contain this Required Action. Therefore, a change is not proposed to the NMP2 TS. TSTF-505 changes are incorporated. The NMP2 BWR/5 design uses wording of two ECCS injection or one ECCS injection and one ECCS spray.
One ADS valve inoperable.	3.5.1.E	3.5.1.E.1	Yes	TSTF-505 changes are incorporated.
One ADS valve inoperable and Condition A entered.	3.5.1.F.1 3.5.1.F.2	3.5.1.F.1 3.5.1.F.2	Yes Yes	TSTF-505 changes are incorporated. TSTF-505 changes are incorporated.
RCIC [Reactor Core Isolation Cooling] System	**3.5.3**	**3.5.3**		
RCIC system inoperable.	3.5.3.A	3.5.3.A.2	Yes	TSTF-505 changes are incorporated.
Primary Containment Air Lock	**3.6.1.2**	**3.6.1.2**		
Primary containment air lock inoperable for reasons other than Condition A or B.	3.6.1.2.C.3	3.6.1.2.C.3	Yes	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one primary containment airlocks inoperable, excessive leakage or a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when leakage exceeds limits or there is a loss of function.

Cross-Reference of TSTF-505 and
Nine Mile Point Nuclear Station, Unit 2 Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	NMP2 Tech Spec	Apply RICT?	Comments
Primary Containment Isolation Valves (PCIVs)	**3.6.1.3**	**3.6.1.3**		
One or more penetration flow paths with one PCIV inoperable except due to leakage not within limit.	3.6.1.3.A.1	3.6.1.3.A.1	Yes	TSTF-505 changes are incorporated.
One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits.	3.6.1.3.E.1	3.6.1.3.E.1	Yes	Note, NMP2 TS 3.6.1.3.E.1 specifies containment purge exhaust valves. TSTF-505 change is incorporated. However, under certain circumstances, with more than one containment purge valve not within leakage limits, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when there is a loss of function.
Residual Heat Removal (RHR) Drywell Spray System	**3.6.1.7**	**3.6.1.6**		NMP2 aligns with the BWR/6, NUREG-1434, for Residual Heat Removal (RHR) Drywell Spray System.
One RHR drywell spray subsystem inoperable.	3.6.1.7.A.1	3.6.1.6.A.1	Yes	TSTF-505 change is incorporated.
Reactor Building-to-Suppression Chamber Vacuum Breakers	**3.6.1.7**	**---------**		The NMP2 TS do not contain this TS. Therefore, a change is not proposed to the NMP2 TS.
Suppression Chamber-to-Drywell Vacuum Breakers	**3.6.1.8**	**3.6.1.7**		

**Cross-Reference of TSTF-505 and
Nine Mile Point Nuclear Station, Unit 2 Technical Specifications**

Tech Spec Description	TSTF-505 Tech Spec	NMP2 Tech Spec	Apply RICT?	Comments
One required suppression chamber-to-drywell vacuum breaker inoperable for opening.	3.6.1.8.A.1	3.6.1.7.A.1	Yes	TSTF-505 changes are incorporated. NMP2 Condition A states "One line with one or more suppression chamber-to-drywell vacuum breakers inoperable for opening." As such, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when loss of function can occur.
Residual Heat Removal (RHR) Suppression Pool Cooling	**3.6.2.3**	**3.6.2.3**		
One RHR suppression pool cooling subsystem inoperable.	3.6.2.3.A.1	3.6.2.3.A.1	Yes	TSTF-505 changes are incorporated.
Residual Heat Removal (RHR) Suppression Pool Spray	**3.6.2.4**	**3.6.2.4**		
One RHR suppression pool spray subsystem inoperable.	3.6.2.4.A.1	3.6.2.4.A.1	Yes	TSTF-505 changes are incorporated.
Drywell Cooling System Fans	**3.6.3.1**	**---------**	No	The NMP2 TS do not contain this TS. Therefore, a change is not proposed to the NMP2 TS.
Residual Heat Removal Service Water (RHRSW) System	**3.7.1**	**---------**	No	The NMP2 TS do not contain this TS. Therefore, a change is not proposed to the NMP2 TS.
[Plant Service Water (PSW)] System and [Ultimate Heat Sink (UHS)]	**3.7.2**	**3.7.1**		NMP2, TS 3.7.1 is the Service Water (SW) and Ultimate Heat Sink (UHS).
One [PSW] pump in each subsystem inoperable.	3.7.2.B	---------	No	The NMP2 TS do not contain this TS. Therefore, a change is not proposed to the NMP2 TS.

Cross-Reference of TSTF-505 and
Nine Mile Point Nuclear Station, Unit 2 Technical Specifications

Tech Spec Description	**TSTF-505 Tech Spec**	**NMP2 Tech Spec**	**Apply RICT?**	**Comments**
One or more cooling towers with one cooling tower fan inoperable.	3.7.2.C	---------	No	The NMP2 TS do not contain this TS. Therefore, a change is not proposed to the NMP2 TS.
---------	---------	3.7.1.A.2	Yes	TSTF-505 changes are incorporated. 3.7.1, Condition A, One SW supply header cross connect valve inoperable, is a NMP2 site specific Condition. Required Action 3.7.1.A.2, Restore the SW supply header cross connect valve to OPERABLE status.
One PSW subsystem inoperable for reasons other than Condition[s] A [and C].	3.7.2.E	3.7.1.C.1	Yes	TSTF-505 changes are incorporated. 3.7.1, Condition C, One SW subsystem inoperable for reasons other than Conditions A and B.
---------	---------	3.7.1.D.1	Yes	TSTF-505 changes are incorporated. 3.7.1, Condition D, One division of intake deicer heaters inoperable.
	---------	3.7.1.E.1	Yes	TSTF-505 changes are incorporated. 3.7.1, Condition E, One required SW pump not in operation and Required Action 3.7.1.E.1, Restore required SW pump to operation are site specific.
	---------	3.7.1.F.1	Yes	TSTF-505 changes are incorporated. 3.7.1, Condition F, Two or more required SW pumps not in operation and Required Action 3.7.1.F.1, Restore all but one required SW pump to operation are site specific. As such, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when loss of function can occur.

Cross-Reference of TSTF-505 and
Nine Mile Point Nuclear Station, Unit 2 Technical Specifications

Tech Spec Description	TSTF-505 Tech Spec	NMP2 Tech Spec	Apply RICT?	Comments
Main Turbine Bypass System	**3.7.7**	**3.7.5**		
Requirements of the LCO not met or Main Turbine Bypass System inoperable.	3.7.7.A	3.7.5.A.1	Yes	TSTF-505 changes are incorporated. SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. PRA success criteria is based on calculations for the minimum valves required to prevent major demands on the suppression pool. The design basis success criteria is based on the bypass system capacity compared with individual TBV.
AC Sources – Operating	**3.8.1**	**3.8.1**		
One [required] offsite circuit inoperable.	3.8.1.A.3	3.8.1.A.3	Yes	TSTF-505 changes are incorporated.
One [required] DG inoperable.	3.8.1.B.4	3.8.1.B.4	Yes	TSTF-505 changes are incorporated.
Two [required] offsite circuits inoperable.	3.8.1.C.2	3.8.1.C.2	Yes	TSTF-505 changes are incorporated.
One [required] offsite circuit inoperable. AND One [required] DG inoperable.	3.8.1.D.1 3.8.1.D.2	3.8.1.D.1 3.8.1.D.2	Yes Yes	TSTF-505 changes are incorporated. TSTF-505 changes are incorporated.
[One [required] [automatic load sequencer] inoperable.	3.8.1.F	---------	No	The NMP2 TS do not contain this TS. Therefore, a change is not proposed to the NMP2 TS.
DC Sources - Operating	**3.8.4**	**3.8.4**		

Cross-Reference of TSTF-505 and
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Tech Spec Description	TSTF-505 Tech Spec	NMP2 Tech Spec	Apply RICT?	Comments
One [or two] battery chargers on one division] inoperable.	3.8.4.A.3	---------	No	The NMP2 TS do not contain this TS. Therefore, a change is not proposed to the NMP2 TS.
One [or two] batter[y][ies on one division] inoperable.	3.8.4.B	---------	No	The NMP2 TS do not contain this TS. Therefore, a change is not proposed to the NMP2 TS.
One DC electrical power subsystems inoperable.	3.8.4.C	3.8.4.A.1	Yes	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.
Inverters - Operating	**3.8.7**	**3.8.7**		
One [required] inverter inoperable.	3.8.7.A	3.8.7.A.1	Yes	TSTF-505 changes are incorporated.
Distribution Systems - Operating	**3.8.9**	**3.8.8**		
One or more AC electrical power distribution subsystems inoperable.	3.8.9.A	3.8.8.A.1	Yes	TSTF-505 changes are incorporated. EDITORIAL: The condition specified in NMP2 TS 3.8.8, Action A.1, is one or both Division 1 and 2 AC electrical power distribution subsystems inoperable. As such, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when loss of function can occur.

Cross-Reference of TSTF-505 and
Nine Mile Point Nuclear Station, Unit 2 Technical Specifications

Tech Spec Description	**TSTF-505 Tech Spec**	**NMP2 Tech Spec**	**Apply RICT?**	**Comments**
One or more AC vital buses inoperable.	3.8.9.B	3.8.8.B.1	Yes	TSTF-505 changes are incorporated. EDITORIAL: The condition specified in NMP2 TS 3.8.8, Action B.1, is one or both Division 1 and 2 120 VAC uninterruptible electrical power distribution subsystems inoperable. As such, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when loss of function can occur.
One or more DC electrical power distribution subsystems inoperable.	3.8.9.C	3.8.8.C.1	Yes	TSTF-505 changes are incorporated. EDITORIAL: The condition specified in NMP2 TS 3.8.8, Action C.1, is one or both Division 1 and 2 DC electrical power distribution subsystems inoperable. As such, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when loss of function can occur.
Programs and Manuals	**5.5**	**5.5**		
Programs and Manuals	5.5.15	[NEW TS] 5.5.15		The NMP2 TS do not currently contain this program. The new RICT Program will be added to the NMP2 TS 5.5.15 consistent with TSTF-505.

ATTACHMENT 5

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

Information Supporting Instrumentation Redundancy and Diversity

Information Supporting Redundancy and Diversity

The following Instrumentation Technical Specifications (TS) Sections are included in this TSTF-505 License Amendment Request (LAR) for Nine Mile Point Nuclear Station, Unit 2 (NMP2).

1. Reactor Protection System (RPS) Instrumentation - TS Section 3.3.1.1
2. Feedwater System and Main Turbine High-Water Level Trip Instrumentation - TS Section 3.3.2.2
3. End-of-Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation - TS Section 3.3.4.1
4. Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation - TS Section 3.3.4.2
5. Emergency Core Cooling System (ECCS) Instrumentation - TS Section 3.3.5.1
6. Reactor Core Isolation Cooling (RCIC) System Instrumentation - TS Section 3.3.5.3
7. Primary Containment Isolation Instrumentation - TS Section 3.3.6.1
8. Mechanical Vacuum Pump Isolation Instrumentation – TS Section 3.3.7.2
9. Loss-of-Power (LOP) Instrumentation - TS Section 3.3.8.1

NMP2 TS Section 3.3 Limiting Conditions for Operation (LCOs) were developed to ensure that NMP2 maintains necessary redundancy and diversity, and complies with the “single failure” design criterion as defined in IEEE-279-1971, and the diversity requirements as defined in Appendix A, “General Design Criteria for Nuclear Power Plants” (GDC), 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.

The following abbreviations are used within the ‘Event’ column of the included tables:

DBA – Design Bases Accident
IMF-AOT – Incident of Moderate Frequency Anticipated Operational Transient
II-AOT – Infrequent Incident – Anticipated Operational Transient
ATWS – Anticipated Transient Without Scram

1. Reactor Protection System (RPS)

Reference: TS 3.3.1.1 Reactor Protection System (RPS) Instrumentation

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

- Each Functional Unit has multiple channels.
- Each Functional Unit will cause a reactor trip with 2/4 tripped channels.
- A failed channel does not cause or prevent a trip.

Diverse inputs trip the reactor (USAR Table 7.2-1).

2/4 is defined as four channels, two trip systems, two channels per trip system arranged in one-out-of-two twice (de-energize to trip) logic, e.g., (Channel A1 or Channel A2) and (Channel B1 or Channel B2).

- Intermediate Range Monitors (Neutron Flux - Upscale) - 2/4
  2 monitors per channel (TS requires only 3 for A1/A2 and 3 for B1/B2)

Information Supporting Redundancy and Diversity

- Average Power Range Monitors - 2/4
    - Neutron Flux (Upscale, Setdown)
      4 voters / channel (two of three APRM votes needed to trip each voter)
    - Flow-Biased Simulated Thermal Power (Upscale)
      4 voters / channel (two of three APRM votes needed to trip each voter)
    - Fixed Neutron Flux (Upscale)
      4 voters / channel (two of three APRM votes needed to trip each voter)
    - Oscillation Power Range Monitor (OPRM) (Upscale)
      4 voters / channel (two of three APRM votes needed to trip each voter)
- Reactor Vessel Steam Dome Pressure (High) - 2/4
- Reactor Vessel Water Level (Low - Level 3) - 2/4
- Main Steam Isolation Valve (Closure) - 2/4
  Channel A1 is 3 or more of the inboard valves
  Channel A2 is 3 or more of the outboard valves
  Channel B1 is 3 or more of the inboard valves
  Channel B2 is 3 or more of the outboard valves
- Drywell Pressure (High) - 2/4
- Scram Discharge Volume Water Level
    - Transmitter / Trip Unit (High) - 2/4
    - Float Switch (High) - 2/4
- Turbine Stop Valve (Closure) - 2/4
  Channel A1 is 3 or more TSVs
  Channel A2 is 3 or more TSVs
  Channel B1 is 3 or more TSVs
  Channel B2 is 3 or more TSVs
- Turbine Control Valve Fast Closure (Trip Oil Pressure - Low) - 2/4
- Reactor Mode Switch (Shutdown Position) - 2/4
  Single Mode Switch with 4 channels
- Manual Scram - 2/4

In addition, NMP2 has a Redundant Reactivity Control System (RRCS) that is designed to provide a redundant and diverse method of shutting down the reactor in the unlikely event that the RPS does not scram the reactor. A signal is transmitted to open the Alternate Rod Insertion (ARI) valves that vent the Control Rod Drive (CRD) scram air header to insert the control rods into the reactor. A signal is also transmitted to the Recirculation Pump Trip (RPT) breakers to trip the reactor recirculation pumps to reduce the reactor power via negative void reactivity feedback.

The NMP2 design also has a Standby Liquid Control System (SLCS) as an independent backup system. A signal is transmitted to initiate the SLCS to inject boron into the Reactor Vessel and to initiate closure of the Reactor Water Clean-Up (RWCU) isolation valves to prevent removal of the injected boron.

Unit 2 RPS Instrumentation Diversity					
TS Table	**RPS Instrument Function**	**Credited Safety Analysis Event**		**Diverse RPS Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.1.1	1. Intermediate Range Monitors				
	a. Neutron Flux - Upscale	15.4.1	Rod Withdrawal Error - Low Power	1) Automatic Initiation – - IRM High Neutron Flux Trip - APRM Upscale, Setdown Trip (most power levels) 2) Manual SCRAM (SCRAM Button or Reactor Mode Switch)	II-AOT
	1b. Inoperable	None (RWM Credited)	None (RWM Credited)	Manual SCRAM	-

3.3.1.1	2. Average Power Range Monitor				
	a. Neutron Flux - Upscale, Setdown	None	None	1) Automatic Initiation – - APRM Upscale, Setdown Trip (most power levels) - IRM High Neutron Flux Trip 2) Manual SCRAM	-
	b. Flow Biased Simulated Thermal Power - Upscale	15.1.1	Loss of Feedwater Heating	1) Automatic Initiation – - APRM Flow Biased Simulated Thermal Power – Upscale - APRM Fixed Neutron Flux – Upscale -TSV Closure Trip 2) Manual SCRAM	II-AOT
	c. Fixed Neutron Flux - Upscale	15.1.6	Inadvertent RHR Shutdown Cooling Operation	1) Automatic Initiation - - APRM Fixed Neutron Flux -Upscale Trip 2) Manual SCRAM	IMF-AOT

Unit 2 RPS Instrumentation Diversity

TS Table	RPS Instrument Function	Credited Safety Analysis Event		Diverse RPS Instrumentation	Event
		USAR Section	Transient / Accident		
		15.2.1	Pressure Regulator Failure - Closed	1) Automatic Initiation - - APRM Fixed Neutron Flux -Upscale Trip - Reactor Vessel Steam Dome Pressure High Trip 2) Manual SCRAM	IMF-AOT
		15.2.4	MSIV Closure Event (Single MSIV Closure)	1) Automatic Initiation - - APRM Fixed Neutron Flux -Upscale Trip - Reactor Vessel Steam Dome Pressure (High) Trip 2) Manual SCRAM	IMF-AOT
		15.4.4	Abnormal Startup of Idle Recirculation Pump	1) Automatic Initiation - - APRM Fixed Neutron Flux -Upscale Trip 2) Manual SCRAM	IMF-AOT
		15.4.5	Recirculation Flow Control Failure with Increasing Flow	1) Automatic Initiation - - APRM Fixed Neutron Flux -Upscale Trip 2) Manual SCRAM	IMF-AOT
		15.4.9	Control Rod Drop Accident	1) Automatic Initiation - - APRM Fixed Neutron Flux -Upscale Trip 2) Manual SCRAM	DBA
	d. Inoperable	None (RBM Credited)	None (RBM Credited)	Manual SCRAM	-
	e. Oscillation Power Range Monitor Upscale	None	None	1) Automatic Initiation - - APRM Oscillation Power Range Monitor Trip 2) Manual SCRAM	-
	f. 2-Out-of-4 Voter	None	None	Manual SCRAM	-

Information Supporting Redundancy and Diversity

Unit 2 RPS Instrumentation Diversity

TS Table	RPS Instrument Function	Credited Safety Analysis Event		Diverse RPS Instrumentation	Event
		USAR Section	Transient / Accident		
3.3.1.1	3. Reactor Vessel Steam Dome Pressure - High	15.2.4	MSIV Closure Event (Single MSIV Closure)	1) Automatic Initiation - - Reactor Vessel Steam Dome Pressure (High) Trip - APRM Fixed Neutron Flux -Upscale Trip 2) Manual SCRAM	IMF-AOT
3.3.1.1	4. Reactor Vessel Level - Low (Level 3)	15.2.7	Loss of Feedwater Flow	1) Automatic Initiation - - Low Water Level (Level 3) Trip - MSIV Closure Trip 2) Manual SCRAM	IMF-AOT
		15.2.8 / 15.6.6	Feedwater Line Break Outside Containment	1) Automatic Initiation - - Low Water Level (Level 3) Trip - MSIV Closure Trip 2) Manual SCRAM	DBA
		15.6.5	Loss of Coolant Accidents	1) Automatic Initiation - Low Water Level (Level 3) Trip - High Drywell Pressure Trip 2) Manual SCRAM	DBA
3.3.1.1	5. Main Steam Isolation Valve - Closure	15.2.4	Main Steam Isolation Valves Closure	1) Automatic Initiation – - MSIV Closure Trip - Reactor Steam Dome Pressure Trip - APRM Fixed Neutron Flux - Upscale Trip 2) Manual SCRAM	IMF-AOT
		15.6.4	Steam System Piping Break Outside Containment	1) Automatic Initiation – - MSIV Closure Trip 2) Manual SCRAM	DBA

Unit 2 RPS Instrumentation Diversity

TS Table	RPS Instrument Function	Credited Safety Analysis Event		Diverse RPS Instrumentation	Event
		USAR Section	Transient / Accident		
3.3.1.1	6. Drywell Pressure - High	15.6.5	Loss of Coolant Accidents	1) Automatic Initiation – - High Drywell Pressure Trip - Reactor Water Level Low Trip 2) Manual SCRAM	DBA
3.3.1.1	7. Scram Discharge Volume Water Level - High				
	a. Transmitter / Trip Unit	15.8.3.6	Scram Discharge Volume	1) Automatic Initiation – - Two diverse sensors volume 2) Manual SCRAM	-
	b. Float Switch	15.8.3.6	Scram Discharge Volume	1) Automatic Initiation – - Two diverse sensors volume 2) Manual SCRAM	-
3.3.1.1	8. Turbine Stop Valve Closure	15.1.2	Feedwater Controller Failure - Maximum Demand	1) Automatic Initiation – - TSV Closure Trip - TCV Closure Trip - Reactor Vessel Steam Dome Pressure High Trip - APRM Fixed Neutron Flux - Upscale Trip 2) Manual SCRAM	IMF-AOT
		15.1.3	Pressure Regulator Failure - Open	1) Automatic Initiation – - TSV Closure Trip - MSIV Closure Trip - Reactor Vessel Steam Dome Pressure High Trip - APRM Fixed Neutron Flux - Upscale Trip 2) Manual SCRAM	IMF-AOT

Unit 2 RPS Instrumentation Diversity					
TS Table	RPS Instrument Function	Credited Safety Analysis Event		Diverse RPS Instrumentation	Event
		USAR Section	Transient / Accident		
		15.2.3	Turbine Trip	1) Automatic Initiation – - TSV Closure Trip - Reactor Vessel Steam Dome Pressure High Trip - APRM Fixed Neutron Flux - Upscale Trip 2) Manual SCRAM	II-AOT
		15.2.5	Loss of Condenser Vacuum	1) Automatic Initiation – - TSV Closure Trip - MSIV Closure Trip - Reactor Vessel Steam Dome Pressure High Trip - APRM Fixed Neutron Flux - Upscale Trip 2) Manual SCRAM	IMF-AOT
		15.3.1	Recirculation Pump Trip	1) Automatic Initiation – - TSV Closure Trip - MSIV Closure Trip - Reactor Vessel Steam Dome Pressure High Trip 2) Manual SCRAM	IMF-AOT
		15.3.2	Recirculation Flow Control Failure - Decreasing Flow	1) Automatic Initiation – - TSV Closure Trip - MSIV Closure Trip - Reactor Vessel Steam Dome Pressure High Trip 2) Manual SCRAM	IMF-AOT

Unit 2 RPS Instrumentation Diversity					
TS Table	**RPS Instrument Function**	**Credited Safety Analysis Event**		**Diverse RPS Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.1.1		15.3.3	Recirculation Pump Seizure	1) Automatic Initiation – - TSV Closure Trip - MSIV Closure Trip - Reactor Vessel Steam Dome Pressure High Trip 2) Manual SCRAM	DBA
		15.3.4	Recirculation Pump Shaft Break	1) Automatic Initiation – - TSV Closure Trip - MSIV Closure Trip - Reactor Vessel Steam Dome Pressure High Trip 2) Manual SCRAM	DBA
3.3.1.1	9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	15.2.2	Generator Load Rejection	1) Automatic Initiation – - Turbine Control Valve Trip - Reactor Vessel Steam Dome Pressure High Trip - APRM Fixed Neutron Flux - Upscale Trip 2) Manual SCRAM	II-AOT
		15.2.6	Loss of AC Power	1) Automatic Initiation – - Turbine Control Valve Trip - Reactor Vessel Steam Dome Pressure High Trip - APRM Fixed Neutron Flux - Upscale Trip 2) Manual SCRAM	IMF-AOT

Unit 2 RPS Instrumentation Diversity					
TS Table	**RPS Instrument Function**	**Credited Safety Analysis Event**		**Diverse RPS Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.1.1	10. Reactor Mode Switch to Shutdown	None	a. Manual SCRAM	Manual Trip - Manual SCRAM Button - De-energize Trip System Power - Manually initiate RRCS	-
3.3.1.1	11. Manual Scram	None	a. Manual SCRAM	Manual Trip - Mode Switch to Shutdown - De-energize Trip System Power - Manually initiate RRCS	-

2. Feedwater System and Main Turbine High-Water Level Trip

Reference: TS 3.3.2.2 Feedwater System and Main Turbine High-Water Level Trip Instrumentation

The Feedwater System and Main Turbine High-Water Level Trip design creates defense-in-depth from the redundancy of the channels for the Trip Function.

- Trip Function has multiple channels.
- Trip Function will cause an Isolation Actuation with 2/3 tripped channels.
- A failed channel does cause or prevent a trip.

2/3 is defined as three channels with one trip system arranged in a two-out-of-three (energize to initiate) logic, e.g., (Channel A and B) or (Channel A and C) or (Channel B and C).

- o Reactor Vessel Water Level (High - Level 8) - 2/3

Unit 2 Feedwater System and Main Turbine High-Water Level Trip Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		USAR Section	Transient / Accident		
3.3.2.2	Reactor Vessel Water Level - High (Level 8)	15.1.1	Loss of Feedwater Heating	1) Automatic Initiation - Reactor Vessel Water Level Turbine Trip / SCRAM - High Turbine Moisture Separator Level Turbine Trip / SCRAM 2) Manual SCRAM	II-AOT
		15.1.2	Feedwater Controller Failure - Maximum Demand	1) Automatic Initiation - Reactor Vessel Water Level Turbine Trip / SCRAM - High Turbine Moisture Separator Level Turbine Trip / SCRAM 2) Manual SCRAM	IMF-AOT
		15.1.3	Pressure Regulator Failure - Open	1) Automatic Initiation - Reactor Vessel Water Level Turbine Trip / SCRAM - High Turbine Moisture Separator Level Turbine Trip / SCRAM 2) Manual SCRAM	IMF-AOT
		15.2.3	Turbine Trip	1) Automatic Initiation - Reactor Vessel Water Level Turbine Trip / SCRAM - High Turbine Moisture Separator Level Turbine Trip / SCRAM 2) Manual SCRAM	II-AOT
		15.2.4	Loss of Condenser Vacuum	1) Automatic Initiation - Reactor Vessel Water Level Turbine Trip / SCRAM - High Turbine Moisture Separator Level Turbine Trip / SCRAM 2) Manual SCRAM	IMF-AOT

Unit 2 Feedwater System and Main Turbine High-Water Level Trip Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		USAR Section	Transient / Accident		
		15.3.1	Recirculation Pump Trip	1) Automatic Initiation - Reactor Vessel Water Level Turbine Trip / SCRAM - High Turbine Moisture Separator Level Turbine Trip / SCRAM 2) Manual SCRAM	IMF-AOT
		15.3.2	Recirculation Flow Control Failure -Decreasing Flow	1) Automatic Initiation - Reactor Vessel Water Level Turbine Trip / SCRAM - High Turbine Moisture Separator Level Turbine Trip / SCRAM 2) Manual SCRAM	IMF-AOT
		15.3.3	Recirculation Pump Seizure	1) Automatic Initiation - Reactor Vessel Water Level Turbine Trip / SCRAM - High Turbine Moisture Separator Level Turbine Trip / SCRAM 2) Manual SCRAM	DBA
		15.3.4	Recirculation Pump Shaft Break	1) Automatic Initiation - Reactor Vessel Water Level Turbine Trip / SCRAM - High Turbine Moisture Separator Level Turbine Trip / SCRAM 2) Manual SCRAM	DBA

3. End-of-Cycle Recirculation Pump Trip (EOC-RPT)

Reference: TS 3.3.4.1 End-of-Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

The EOC-RPT design creates defense-in-depth from the redundancy of the channels for the Trip Function.

- Trip Function has multiple channels.
- Trip Function will cause an Actuation with 2/2 tripped channels.
- A failed channel does cause or prevent a trip.

2/2 is defined as four channels, two trip systems, two channels per trip system arranged in a two-out-of-two once (energize to trip) logic, e.g., (Channel A and Channel B) or (Channel C and Channel D).

  - Turbine Stop Valve (TSV) (Closure) - 2/2
  - Turbine Control Valve (TCV) Fast Closure (Trip Oil Pressure - Low) - 2/2

Unit 2 End-of-Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		USAR Section	Transient / Accident		
3.3.4.1	Turbine Stop Valve Closure	15.1.1	Loss of Feedwater Heating	1) Automatic Initiation - TSV Closure Trip 2) Manual Trip	II-AOT
		15.1.2	Feedwater Controller Failure - Maximum Demand	1) Automatic Initiation - TSV Closure Trip 2) Manual Trip	IMF-AOT
		15.1.3	Pressure Regulator Failure - Open	1) Automatic Initiation - TSV Closure Trip 2) Manual Trip	IMF-AOT
		15.2.3	Turbine Trip	1) Automatic Initiation - TSV Closure Trip 2) Manual Trip	II-AOT
		15.2.5	Loss of Condenser Vacuum	1) Automatic Initiation - TSV Closure Trip 2) Manual Trip	IMF-AOT
		15.3.2	Recirculation Flow Control Failure - Decreasing Flow	1) Automatic Initiation - TSV Closure Trip 2) Manual Trip	IMF-AOT
		15.3.3	Recirculation Pump Seizure	1) Automatic Initiation - TSV Closure Trip 2) Manual Trip	DBA
3.3.4.1	Turbine Control Valve Fast Closure	15.2.2	Generator Load Rejection	1) Automatic Initiation - TCV Closure Trip 2) Manual Trip	II-AOT
		15.2.6	Loss of AC Power	1) Automatic Initiation - TCV Closure Trip 2) Manual Trip	IMF-AOT

4. Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT)

Reference: TS 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

The ATWS-RPT design creates defense-in-depth from the redundancy of the channels for the Trip Function.

- Trip Function has multiple channels.
- Trip Function will cause an Actuation with 2/2 tripped channels.
- A failed channel does cause or prevent a trip.

2/2 is defined as four channels, two trip systems, two channels per trip system arranged in a two-out-of-two once (energize to trip) logic, e.g., (Channel 1A and Channel 1B) or (Channel 2A and Channel 2B).

- o Reactor Vessel Water Level (Low Low - Level 2) - 2/2
- o Reactor Vessel Steam Dome Pressure (High) - 2/2

Unit 2 Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS-RPT) Instrumentation Diversity					
TS Table	**Instrument Function**	**Credited Safety Analysis Event**		**Diverse Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.4.2	Reactor Vessel Water Level Low Low (Level 2)	15.8	Anticipated Transient Without SCRAM	1. Automatic Initiation - -ATWS-RPT (If APRM's not downscale) 2. Manual RPT	ATWS
3.3.4.2	Reactor Vessel Steam Dome Pressure - High	15.8	Anticipated Transient Without SCRAM	1. Automatic Initiation - -ATWS-RPT (If APRM's not downscale) 2. Manual RPT	ATWS

5. Reactor Core Isolation Cooling (RCIC)

Reference: TS 3.3.5.3 Reactor Core Isolation Cooling (RCIC) System Instrumentation

The RCIC design creates defense-in-depth because of the redundancy of the channels for the Initiation Function.

- Initiation Function has multiple channels.
- Initiation Function will cause an actuation with 1/2 and 2/4 tripped channels.
- A failed channel does not cause or prevent an initiation.

2/4 is defined as four channels, one trip system arranged in a one-out-of-two twice (energize to initiate) logic, e.g., (Channel A or Channel C) and (Channel E or Channel G).

1/2 is defined as two channels, one trip system arranged in a one-out-of-two once (energize to initiate) logic, e.g., (Channel A) or (Channel C).

  - Reactor Vessel Water Level (Low Low - Level 2) - 2/4
  - Reactor Vessel Water Level (High - Level 8) - 2/4
  - Pump Suction Pressure (Low) - 1/2
  - Manual Initiation - 1/system - one trip system with one channel

Unit 2 Reactor Core Isolation Cooling Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		USAR Section	Transient / Accident		
3.3.5.3	Reactor Vessel Water Level Low Low (Level 2)	15.1.2	Feedwater Controller Failure - Maximum Demand	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	IMF-AOT
		15.1.3	Pressure Regulator Failure - Open	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	IMF-AOT
		15.2.6	Loss of AC Power	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	IMF-AOT
		15.2.7	Loss of Feedwater Flow	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	IMF-AOT
		15.3.1	Recirculation Pump Trip	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	IMF-AOT
		15.3.2	Recirculation Flow Control Failure -Decreasing Flow	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	IMF-AOT
		15.3.3	Recirculation Pump Seizure	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	DBA
		15.2.8 / 15.6.6	Feedwater Line Break Outside Containment	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	DBA
3.3.5.3	Reactor Vessel Water Level High (Level 8)	None	None	1. Automatic RCIC Stop - High Level (L8) 2. Manually Secure	-
3.3.5.3	Pump Suction Pressure - Low	None	None	1. Automatically Initiated (Swap Suction Source) - Suction Pressure Low 2. Manually Swap Suction Sources	-

6. Emergency Core Cooling System (ECCS)

Reference: TS 3.3.5.1 Emergency Core Cooling System Instrumentation

The ECCS design creates defense-in-depth from the redundancy of the channels for the Trip Function (ECCS System Actuation)

- Trip Function (ECCS Actuation) has multiple channels.
- Trip Function (ECCS Actuation) will cause a Trip Function with 2/2, 4/4, 2/4 or 1/2 tripped channels.
- A failed channel does cause or prevent a trip (except Manual).

2/2 is defined as two channels, one trip system arranged in a two-out-of-two once (energize to initiate) logic, e.g., (Channel A and Channel B).

4/4 is defined as four channels, one trip system arranged in a four-out-of-four (energize to initiate) logic, e.g., (Channel A and B and C and D).

2/4 is defined as four channels, one trip system arranged in a one-out-of-two twice (energize to initiate) logic, e.g., (Channel A or Channel C) and (Channel B or Channel D).

1/2 is defined as two channels, one trip system arranged in a one-out-of-two once (energize to initiate) logic, e.g., Channel A or Channel B).

- Low Pressure Coolant Injection (LPCI) - Loop A and Low Pressure Core Spray (LPCS) (LPCI - A and LPCS share a common logic)
  - Reactor Vessel Water Level (Low Low Low - Level 1) - 2/4
    2 reactor vessel water level channels are combined with 2 drywell pressure channels
  - Drywell Pressure (High) - 2/4
    2 drywell pressure channels are combined with 2 reactor vessel water level channels
  - Manual Initiation - 2/2
- Low Pressure Coolant Injection (LPCI) - Loop B and Loop C
  (LPCI - B and LPCI - C share a common logic)
  - Reactor Vessel Water Level (Low Low Low - Level 1) - 2/4
    2 reactor vessel water level channels are combined with 2 drywell pressure channels
  - Drywell Pressure (High) - 2/4
    2 drywell pressure channels are combined with 2 reactor vessel water level channels
  - Manual Initiation - 2/2
- High Pressure Core Spray (HPCS) System
  - Reactor Vessel Water Level (Low Low - Level 2) - 2/4
  - Drywell Pressure (High) - 2/4
  - Manual Initiation - 2/2
- Automatic Depressurization System (ADS) 2 Systems - A & B
  Either system A or B will cause all ADS valves to open.
  - Reactor Vessel Water Level (Low Low Low - Level 1) - 2/2
  - ADS Initiation Timer - 1/1
  - Reactor Vessel Water Level (Low - Level 3 - Permissive) - 1/1
  - LPCI-A and LPCS Pump Discharge Pressure (High) - 2/2
    (System A only)
  - LPCI-B and LPCS-C Discharge Pressure (High) - 2/2
    (System B only)
  - Manual Initiation - 4/4

  All contacts in one trip system must close to initiate ADS trip system.

Unit 2 Emergency Core Cooling System (ECCS) Instrumentation Diversity					
TS Table	**Instrument Function**	**Credited Safety Analysis Event**		**Diverse Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.5.1-1	1. Low Pressure Coolant Injection-A (LPCI) and Low Pressure Core Spray (LPCS) Subsystems				
	Reactor Vessel Water Level Low Low Low, Level 1	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation - RX Water Vessel Low Low Low + LPCI / LPCS Time Delay Relay (Normal Or Emergency Power) + LPCI / LPCS Discharge Pressure Low 2. Manual Injection / Spray	DBA
	Drywell Pressure - High	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation - Drywell Pressure High + LPCI / LPCS Time Delay Relay (Normal Or Emergency Power) + LPCI / LPCS Discharge Pressure Low 2. Manual Injection / Spray	DBA
	Manual Initiation	15.2.9	Failure of RHR Shutdown Cooling	Manual Initiation	IMF-AOT
3.3.5.1-1	2. LPCI B and LPCI C Subsystems				
	Reactor Vessel Water Level Low Low Low, Level 1	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation - RX Water Vessel Low Low Low + LPCI / LPCS Time Delay Relay (Normal Or Emergency Power) + LPCI / LPCS Discharge Pressure Low 2. Manual Injection / Spray	DBA
	Drywell Pressure - High	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation - Drywell Pressure High + LPCI / LPCS Time Delay Relay (Normal Or Emergency Power) + LPCI / LPCS Discharge Pressure Low 2. Manual Injection / Spray	DBA
	Manual Initiation	15.2.9	Failure of RHR Shutdown Cooling	Manual Initiation	DBA

Unit 2 Emergency Core Cooling System (ECCS) Instrumentation Diversity					
TS Table	**Instrument Function**	**Credited Safety Analysis Event**		**Diverse Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.5.1-1	3. High Pressure Core Spray (HPCS) System				
	Reactor Vessel Water Level - Low Low, Level 2	15.1.3	Pressure Regulator Failure - Open	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	IMF-AOT
		15.2.6	Loss of AC Power	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	IMF-AOT
		15.2.8 / 15.6.6	Feedwater Line Break Outside Containment	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	DBA
		15.2.7	Loss of Feedwater Flow	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	IMF-AOT
		15.3.1	Recirculation Pump Trip	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	IMF-AOT
		15.3.2	Recirculation Flow Control Failure -Decreasing Flow	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	IMF-AOT
		15.3.3	Recirculation Pump Seizure	1. Automatic Initiation - RX Water Level Low Low 2. Manual Initiation	DBA
	Drywell Pressure - High	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation - Drywell Pressure High 2. Manual Initiation	DBA

Unit 2 Emergency Core Cooling System (ECCS) Instrumentation Diversity					
TS Table	**Instrument Function**	**Credited Safety Analysis Event**		**Diverse Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.5.1-1	4. Automatic Depressurization System (ADS) Trip System A				
	Reactor Vessel Water Level - Low, Low Low, Level 1	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2. Manual Initiation	DBA
	ADS Initiation Timer	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2. Manual Initiation	DBA
	Reactor Vessel Water Level - Low, Level 3 (Permissive)	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2. Manual Initiation	DBA
	LPCS Pump Discharge Pressure - High	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2. Manual Initiation	DBA
	LPCI Pump A Discharge Pressure - High	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2. Manual Initiation	DBA
	Manual Initiation	15.2.9	Failure of RHR Shutdown Cooling	Manual Initiation	IMF-AOT

Unit 2 Emergency Core Cooling System (ECCS) Instrumentation Diversity					
TS Table	Instrument Function	Credited Safety Analysis Event		Diverse Instrumentation	Event
		USAR Section	Transient / Accident		
3.3.5.1-1	4. Automatic Depressurization System (ADS) Trip System B				
	Reactor Vessel Water Level - Low, Low Low, Level 1	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2. Manual Initiation	DBA
	ADS Initiation Timer	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2. Manual Initiation	DBA
	Reactor Vessel Water Level - Low, Level 3 (Permissive)	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2. Manual Initiation	DBA
	LPCS Pump Discharge Pressure - High	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2. Manual Initiation	DBA
	LPCI Pump A Discharge Pressure - High	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation -RX Water Level Low, Low, Low + ADS Initiation Timer + RX Water Level Low + LPCS & LPCI Discharge Pressure – High 2. Manual Initiation	DBA
	Manual Initiation	15.2.9	Failure of RHR Shutdown Cooling	Manual Initiation	IMF-AOT

7. Primary Containment Isolation Instrumentation

Reference: TS 3.3.6.1 Primary Containment Isolation Instrumentation

The isolation actuation design creates defense-in-depth from the redundancy of the channels for each Trip Function.

- Each Trip Function has multiple channels
- Each Trip Function will cause an isolation actuation with 2/2, 2/4, or 1/1 tripped channels.
- A failed channel does not prevent a trip, but may cause a trip depending on the logic design

Diverse inputs for Isolation Actuation (USAR Table 7.3-5)

2/2 is defined as four channels, two trip systems, two channels per trip system arranged in a two-out-of-two once (de-energize to trip) logic, e.g., (Channel A and Channel B) or (Channel C and Channel D).

2/4 is defined as four channels, two trip systems, two channels per trip system arranged in a one-out-of-two twice (de-energize to trip) logic, e.g., (Channel A or Channel C) and (Channel C or Channel D).

1/1 is defined as two channels, two trip systems, one channel per trip system arranged in a one-out-of-one once (de-energize to trip) logic, e.g., (Channel A) or (Channel D).

- Main Steam Line Isolation
  - Reactor Vessel Water Level (Low Low Low - Level 1) - 2/4*
  - Main Steam Line Pressure (Low) - 2/4*
  - Main Steam Line Flow (High) - This instrumentation uses 16 flow channels, four for each steam line. One channel from each steam line inputs to one of four trip strings. Two trip strings make up each trip system, and both trip systems must trip to cause an MSL isolation. Each trip string has four inputs (one per MSL), any one of which will trip the trip string. The trip strings within a trip system are arranged in a one-out-of-two logic. Therefore, this is effectively a one-out-of-eight taken twice logic arrangement to initiate isolation of the MSIVs.**
  - Condenser Vacuum (Low) - 2/4*
  - Main Steam Line Tunnel Temperature (High) - 2/4*
  - Main Steam Line Tunnel Differential Temperature (High) - 2/4*
  - Main Steam Line Tunnel Lead Enclosure Temperature (High) - This instrumentation uses 12 temperature channels, three for each trip string. One sensor in each trip string is measuring the temperature in one of the three areas of the lead enclosure. Two trip strings make up each trip system, and both trip systems must trip to cause a MSL isolation. One sensor is required to trip the trip string. The trip strings within a trip system are arranged in a one-out-of-two taken twice logic arrangement for each enclosure area to initiate isolation of the MSIVs.***
  - Manual Initiation - 2/4****

    * The outputs from the same channels are arranged into two 2/2 trip systems to isolate all MSL drain valves. One 2/2 trip system is associated with the inboard valve and the other 2/2 trip system is associated with the outboard valves.

    ** The outputs from the 16 flow channels are connected into two two-out-of-two trip systems (effectively two one-out-of-four taken twice logic), with one trip system

isolating the inboard MSL drain valve and the other trip system isolating the outboard MSL drain valves.

*** The 12 temperature channels are connected into two two-out-of-two trip systems for each enclosure area, with one trip system isolating the inboard MSL drain valve and the other trip system isolating the outboard MSL drain valves.

**** To close the MSL drain valves, all channels in both trip system must actuate (i.e., both channels from each of the two associated switch and push buttons are required to actuate the inboard valve trip system and both channels from each of the two associated switch and push buttons are required to actuate the outboard valve trip system).

- Primary Containment Isolation
  - Reactor Vessel Water Level (Low Low - Level 2) - 2/2
  - Drywell Pressure (High) - 2/2
  - Standby Gas Treatment (SGT) System Exhaust Radiation (High) - 1/1
  - Manual Initiation - This function uses eight channels, two per switch and push button. The four channels from two switch and push buttons input into one trip system and the four channels from the other two switch and push buttons input into the other trip system, with the channels connected in a four-out-of-four logic.
- Reactor Core Isolation Cooling (RCIC) System Isolation
  - RCIC Steam Line Flow (High) - 1/1
  - RCIC Steam Supply Pressure (Low) - 2/2
  - RCIC Turbine Exhaust Diaphragm Pressure (High) - 2/2
  - RCIC Equipment Room Area Temperature (High) - 1/1
  - RCIC Steam Line Tunnel Temperature (High) - 1/1
  - RHR Equipment Room Area Temperature (High) - 1/1 with 4 channels
  - Reactor Building Pipe Chase Area Temperature (High) - 1/1 with 8 channels
  - Reactor Building General Area Temperature (High) - 1/1 with 10 channels
  - RCIC/RHR Steam Flow (High) - 1/1
  - Manual initiation - 1/1 with 1 channel
- Reactor Water Cleanup (RWCU) System Isolation
  - Differential Flow (High) - 1/1
  - Heat Exchanger Room Temperature (High) - 1/1
  - Pump Room Temperature (High) - 1/1 with 4 channels
  - Reactor Building Pipe Chase Area Temperature (High) - 1/1 with 8 channels
  - Reactor Vessel Water Level (Low Low - Level 2) - 2/2
  - SLC System Initiation - 1/1 - one channel from each of two pumps
  - Manual Initiation - 2/4
- RHR SDC System Isolation
  - RHR Equipment Room Area Temperature (High) - 1/1 with 4 channels
  - Reactor Vessel Water Level (Low - Level 3) - 2/2
  - Reactor Vessel Pressure (high) - two one-out-of-two
  - Reactor Building Pipe Chase Area Temperature (High) - 1/1 with 8 channels
  - Reactor Building General Area Temperature (High) - 1/1 with 10 channels
  - Manual Initiation - 2/4

Information Supporting Instrumentation Redundancy and Diversity

Unit 2 Primary Containment Isolation Instrumentation Diversity					
TS Table	**Instrument Function**	**Credited Safety Analysis Event**		**Diverse Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.6.1-1	Main Steam Line Isolation				
	Reactor Vessel Water Level - Low Low Low, Level 1	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RX Water Level Low Low Low 2. Manual Isolation	DBA
		15.6.5	Loss of Coolant Accidents	1. Automatic Initiation - RX Water Level Low Low Low 2. Manual Isolation	DBA
	Main Steam Line Pressure - Low	15.1.3	Pressure Regulator Failure - Open	1. Automatic Initiation - MSL Pressure Low 2. Manual Isolation	IMF-AOT
	Main Steam Flow - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - MSL Flow High 2. Manual Isolation	DBA
	Condenser Vacuum - Low	15.2.4	Loss of Condenser Vacuum	1. Automatic Initiation - Condenser Vacuum Low 2. Manual Isolation	DBA
	Main Steam Tunnel Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - MSL Tunnel Temp 2. Manual Isolation	DBA
	Main Steam Tunnel Differential Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - MSL Differential Temp High 2. Manual Isolation	DBA
	Main Steam Line Tunnel Lead Enclosure Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - MSL Tunnel Lead Temperature High 2. Manual Isolation	DBA

Information Supporting Instrumentation Redundancy and Diversity

Unit 2 Primary Containment Isolation Instrumentation Diversity					
TS Table	**Instrument Function**	**Credited Safety Analysis Event**		**Diverse Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.6.1-1	Primary Containment Isolation				
	Reactor Vessel Water Level - Low Low (Level 2)	15.2.7	Loss of Feedwater Flow	1. Automatic Initiation - RX Water Level Low Low 2. Manual Isolation	IMF-AOT
		15.2.8 / 15.6.6	Feedwater Line Break Outside Containment	1. Automatic Initiation - RX Water Level Low Low 2. Manual Isolation	DBA
		15.3.2	Recirculation Flow Control Failure - Decreasing Flow	1. Automatic Initiation - RX Water Level Low Low 2. Manual Isolation	IMF-AOT
		15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RX Water Level Low Low 2. Manual Isolation	DBA
		15.6.5	Loss of Coolant Accidents	1. Automatic Initiation - RX Water Level Low Low 2. Manual Isolation	DBA
	Drywell Pressure - High	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation - Drywell Pressure High 2. Manual Isolation	DBA
	Standby Gas Treatment System Exhaust Radiation - High	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation - SGT Exhaust Radiation High 2. Manual Isolation	DBA
3.3.6.1-1	RCIC Isolation				
	RCIC Steam Flow - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RCIC Steam Flow High + RCIC Steam Line Flow Timer 2. Manual Isolation	DBA
	RCIC Steam Supply Pressure - Low	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RCIC Steam Supply Pressure Low 2. Manual Isolation	DBA

Information Supporting Instrumentation Redundancy and Diversity

Unit 2 Primary Containment Isolation Instrumentation Diversity					
TS Table	**Instrument Function**	**Credited Safety Analysis Event**		**Diverse Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
	RCIC Turbine Exhaust Diaphragm Pressure - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic I Initiation - RCIC Turbine Exhaust Diaphragm Pressure High 2. Manual Isolation	DBA
	RCIC Equipment Room Area Temperature High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RCIC Area Temperature High + Area Temperature Time Delay 2. Manual Isolation	DBA
	RHR Equipment Room Area Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RHR Equipment Room Area Temperature High +Area Temperature Time Delay 2. Manual Isolation	DBA
	Reactor Building Pipe Chase Area Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RB Pipe Chase Area Temperature High +Area Temperature Time Delay 2. Manual Isolation	DBA
	Reactor Building General Area Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RB General Area Temperature High +Area Temperature Time Delay 2. Manual Isolation	DBA
	RCIC / RHR Steam Flow - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RCIC / RHR Steam Flow High 2. Manual Isolation	DBA

Information Supporting Instrumentation Redundancy and Diversity

Unit 2 Primary Containment Isolation Instrumentation Diversity					
TS Table	**Instrument Function**	**Credited Safety Analysis Event**		**Diverse Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.6.1-1	Reactor Water Cleanup Isolation				
	Reactor Vessel Water Level - Low Low (Level 2)	15.6.5	Loss of Coolant Accidents	1. Automatic Initiation - RX Water Level Low Low 2. Manual Isolation	DBA
	Reactor Water Cleanup Differential Flow - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RWCU Differential Flow High + Differential Flow Timer 2. Manual Isolation	DBA
	RWCU Heat Exchanger Area Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RWCU Heat Exchanger Room Area Temperature High +Area Temperature Time Delay 2. Manual Isolation	DBA
	Pump Room Area Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - Pump Room Area Temp High +Area Temperature Time Delay 2. Manual Isolation	DBA
	Reactor Building Pipe Chase Area Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RB Pipe Chase Area Temperature High +Area Temperature Time Delay 2. Manual Isolation	DBA
	SLC System Initiation Signal	15.8.3	ATWS	1. Automatic Initiation - SLS Initiation Signal 2. Manual Isolation	ATWS

Information Supporting Instrumentation Redundancy and Diversity

Unit 2 Primary Containment Isolation Instrumentation Diversity					
TS Table	**Instrument Function**	**Credited Safety Analysis Event**		**Diverse Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.6.1-1	RHR Shutdown Cooling Isolation				
	RHR Equipment Room Area Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RHR Equipment Room Area Temperature High +Area Temperature Time Delay 2. Manual Isolation	DBA
	Reactor Vessel Water Level - Low (Level 3)	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RX Water Level Low 2. Manual Isolation	DBA
	Reactor Vessel Pressure High	None	None	1. Automatic Initiation - RX Vessel Pressure High 2. Manual Isolation	-
	Reactor Building Pipe Chase Area Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RB Pipe Chase Area Temperature High +Area Temperature Time Delay 2. Manual Isolation	DBA
	Reactor Building General Area Temperature - High	15.6.4	Steam System Piping Break Outside Containment	1. Automatic Initiation - RB General Area Temperature High +Area Temperature Time Delay 2. Manual Isolation	DBA

8. Mechanical Vacuum Pump Isolation Instrumentation

Reference: TS 3.3.7.2 Mechanical Vacuum Pump Isolation Instrumentation

The Mechanical Vacuum Pump Isolation Instrumentation design creates defense-in-depth from the redundancy of the channels for the Trip Function.

- Trip Function has multiple channels.
- Trip Fu
- nction will cause an Isolation Actuation with 2/4 tripped channels.
- A failed channel does cause or prevent a trip.

2/4 is defined as four channels, two trip systems, two channels per trip system arranged in one-out-of-two twice (de-energize to trip) logic, e.g., (Channel A1 or Channel A2) and (Channel B1 or Channel B2).

  - Main Steam Line Radiation - High - 2/4

Information Supporting Instrumentation Redundancy and Diversity

Unit 2 Primary Mechanical Vacuum Pump Isolation Instrumentation Diversity					
TS	**Instrument Function**	**Credited Safety Analysis Event**		**Diverse Instrumentation**	**Event**
		USAR Section	**Transient / Accident**		
3.3.7.2	Mechanical Vacuum Pump Isolation				
	Main Steam Line Radiation - High	15.4.9	Control Rod Drop Accident	1. Automatic Initiation - Main Steam Line Radiation - High 2. Manual Initiation	DBA

Information Supporting Instrumentation Redundancy and Diversity

9. Loss-of-Power (LOP)

Reference: TS 3.3.8.1 Loss-of-Power (LOP) Instrumentation

The LOP design creates defense-in-depth from the redundancy of the channels for the Initiation Function.

- A failed channel does not cause or prevent an initiation.

The voltage for the Division 1, 2, and 3 buses is monitored at two levels, which can be considered as two different undervoltage functions: Loss of Voltage and Degraded Voltage.

Accident analyses credit the loading of two of the three DGs based on the loss-of-offsite power coincident with a LOCA. Consequently, three channels of Division 1, 2, and 3 4.16 kV Emergency Bus Loss of Voltage or Degraded Voltage are available, but only two channels are required.

Each 4.16 kV emergency bus has its own independent LOP instrumentation and associated trip logic. The Division 1, 2, and 3, 4.16 kV Emergency Bus Loss of Voltage Function and Degraded Voltage Function is monitored by three separate relays, one relay per phase. These relay outputs are arranged in a two-out-of-three logic configuration for each division. These requirements ensure that no single instrument failure can preclude the DG function (Since a failure of a required degraded voltage channel or a time delay relay channel will only impact the ability of one of the three DGs to start, and only two DGs are credited in the accident analyses, the DG function is still maintained).

Information Supporting Instrumentation Redundancy and Diversity

Regulatory Guide 1.174, Revision 2, Section 2.1.1 – Defense-in-Depth

Defense-in-depth consists of several elements and consistency with the defense-in-depth philosophy is maintained if the following occurs:

- **A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.**
  - Current Technical Specifications (TS) reflect this balance by allowing one sensor module or channel to be placed in trip, while preserving the fundamental safety function of the applicable system. Tripping an inoperable channel does not affect the number of channels required to provide the safety function. Even in the TS condition for two channels in a function inoperable, the fundamental safety function is preserved since sufficient operable channels remain in the function.
- **Over-reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided.**
  - No programmatic activities are relied upon as compensatory measures when one or two channels of the applicable instrumentation are inoperable. The remaining operable channels for that function are fully capable of performing the safety function of the applicable system.
- **System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).**
  - System redundancy, independence and diversity remain the same as in the as-designed condition. The number of operable functions has not been decreased (diversity), the number of minimum operable channels to perform the safety function has not been decreased, and the channels remain independent as originally designed, even with one channel inoperable.
- **Defenses against potential common-cause failures are preserved, and the potential for the introduction of new common-cause failure mechanisms is assessed.**
  - This LAR does not impact the original determination of common-cause failure for the applicable instrumentation and its functions. It may allow the allowed outage time to be extended for one or two channels in a function to be inoperable prior to placing the channel in trip. Placing the channel in trip fulfils one of the two required channels in trip needed to perform the safety function.
- **Independence of barriers is not degraded.**
  - Barriers are not affected by this LAR request.

Information Supporting Instrumentation Redundancy and Diversity

- **Defenses against human errors are preserved.**
  - In the conditions listed in the TS, a potential extension of the allowed outage time does not change any personnel actions required when the TS Action is entered. Therefore, no change to the possibility of a human error is introduced and no change to the defenses against that potential human error have been altered.

- **The intent of the plant's design criteria is maintained.**
  - The design criteria of the applicable systems are maintained as reflected in the Updated Safety Analysis Report (USAR). Redundancy, diversity of signal and independence of trip channel functions are maintained with the requested change. The change requested in the LAR does not physically change the applicable systems in any way. It only allows additional time, under certain low risk conditions in accordance with the Risk Informed Completion Time (RICT) Program, to perform actions that the NRC has previously determined to be acceptable.

Therefore, the defense-in-depth principals prescribed in Regulatory Guide 1.174, Revision 2, are met.

ATTACHMENT 6

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

<u>RICT Program Implementation Items</u>

RICT Program Implementation Items

The table below identifies the items that are required to be completed prior to implementation of the Risk-Informed Completion Time (RICT) Program at Nine Mile Point Unit 2. All issues identified below will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa -2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to the implementation of the RICT Program.

Source	Description	Implementation Item
Enclosure 1, Table E1-1, TS 3.3.7.2.A	Mechanical Vacuum Pump Isolation Instrumentation - One or more channels inoperable	SSCs are not modeled. The model will be updated to include these SSCs prior to exercising the RICT program for this TS. The PRA Success Criteria will match the Design Success Criteria.
Enclosure 1, Table E1-1, TS 3.6.1.7.A	One line with one or more suppression chamber-to-drywell vacuum breakers inoperable for opening.	The model will be updated to include this failure mode prior to exercising the RICT program for this TS. The PRA Success Criteria will match the Design Success Criteria.
Enclosure 1, Table E1-1, TS 3.7.1.C	One SW subsystem inoperable for reasons other than Conditions A and B.	The success criteria are consistent with the design basis except when UHS temperature is > 82°F. The model is being updated to include this condition prior to exercising the RICT program for this TS
Enclosure 1, Table E1-1, TS 3.7.1.D	One division of intake deicer heaters inoperable.	The intake deicer heaters are not directly modeled in the PRA. The model will be updated to explicitly include these components prior to its use with RICT for this TS.
Internal Flooding Analysis	Frequency of floods of various magnitudes in the internal flooding analysis	The NMP2 PRA model will be updated to incorporate the new pipe rupture frequencies using the pipe length approach per the latest revision of EPRI TR-1013141.

Information Supporting Instrumentation Redundancy and Diversity

Source	Description	Implementation Item
Enclosure 2, Table E-2-1, Jensen Hughes Report 032405-RPT-01, April 2019	Nine Mile Point Unit 2 PRA Fact and Observation Independent Assessment & Focused Scope Peer Review	All open issues identified in Report 032405-RPT-01 will be addressed prior to exercising the RICT program.

ATTACHMENT 7

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

<u>Proposed Renewed Facility Operating License Changes (Mark-ups)</u>

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(25) Within 14 days of the license transfers, Exelon Generation shall submit to the NRC the Nuclear Operating Services Agreement reflecting the terms set forth in the application dated August 6, 2013. Section 7.1 of the Nuclear Operating Services Agreement may not be modified in any material respect related to financial arrangements that would adversely impact the ability of the licensee to fund safety-related activities authorized by the license without the prior written consent of the Director of the Office of Nuclear Reactor Regulation.

(26) Within 10 days of the license transfers, Exelon Generation shall submit to the NRC the amended CENG Operating Agreement reflecting the terms set forth in the application dated August 6, 2013. The amended and restated Operating Agreement may not be modified in any material respect concerning decision making authority over safety, security and reliability without the prior written consent of the Director of the Office of Nuclear Reactor Regulation.

(27) At least half the members of the CENG Board of Directors must be U.S. citizens.

(28) The CENG Chief Executive Officer, Chief Nuclear Officer, and Chairman of the CENG Board of Directors must be U.S. citizens. These individuals shall have the responsibility and exclusive authority to ensure and shall ensure that the business and activities of CENG with respect to the facility's license are at all times conducted in a manner consistent with the public health and safety and common defense and security of the United States.

(29)

(30)

Insert 1

Proposed Renewed Facility Operating License Changes (Mark-ups)

INSERT 1

(29) Adoption of Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extension Completion Times – RITSTF Initiative 4b"

Exelon is approved to implement TSTF-505, Revision 2, modifying the Technical Specification requirements related to Completion Times (CT) for Required Actions to provide the option to calculate a longer, risk-informed CT (RICT). The methodology for using the new Risk-Informed Completion Time Program is described in NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, which was approved by the NRC on May 17, 2007.

Exelon will complete the implementation items listed in Attachment 6 of Exelon Letter to the NRC dated October 31, 2019, prior to implementation of the RICT Program. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa -2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to the implementation of the RICT Program.

ENCLOSURE 1

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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. Introduction

Section 4.0, Item 2 of the NRC Final Safety Evaluation (Reference 1 of this Enclosure) for NEI 06-09, 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 ECCS flowrates) for each of the TS requirements, or an appropriate disposition or programmatic restriction will be provided.

This enclosure provides confirmation that the Nine Mile Point Nuclear Station Unit 2 (NMP2) 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 comparison includes each of the TS LCO conditions and associated required actions within the scope of the RICT Program. The NMP2 PRA model has the capability to model directly or through use of a bounding surrogate the risk impact of entering each of the TS LCOs in 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 "Proposed TS LCO Condition": Lists all of the LCOs and condition statements within the scope of the RICT Program.
- Column "SSCs Covered by TS LCO Condition": The SSCs addressed by each action requirement.
- Column "SSCs 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": A summary of the required functions from the design basis analyses.
- Column "Design Success Criteria": A summary of the success criteria from the design basis analyses.
- Column "PRA Success Criteria": The function success criteria modeled in the PRA.
- Column " Other 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, Revision 0-A.

List of Revised Required Actions to Corresponding PRA Functions

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 Real-Time Risk (RTR) tool for the RICT program. Differences in success criteria typically arise due to the requirement in the PRA standard 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 PRA standard as required by NEI 06-09, Revision 0-A.

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). Following 4b implementation, 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, Revision 0-A and the NRC safety evaluation, and may differ from the RICTs presented.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.1.7.A	One SLC subsystem inoperable.	2 SLC trains	Yes	Provide a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown	One of two trains	Two of two trains	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The PRA success criteria are more restrictive than the design basis since both trains are required.
3.3.1.1.A	RPS Instrumentation - One or more required channels inoperable.	Instrumentation outlined in table 3.3.1.1-1 (see Note 1)	Not explicitly	Provide reactor trip signal based on plant parameters	One of two channels, taken twice	Same	Electrical failure of the SCRAM system will be used as a conservative surrogate for failure of any channel of RPS.
3.3.1.1.B	RPS Instrumentation - One or more Functions with one or more required channels inoperable in both trip systems.	See 3.3.1.1.A					

List of Revised Required Actions to Corresponding PRA Functions

Table E1-1: In Scope TS/LCO Conditions to Corresponding PRA Functions							
Technical Specification (TS)	**TS Condition Description**	**SSCs Covered by TS LCO Condition**	**SSCs Modeled in PRA?**	**Function Required by TS LCO Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Other Comments**
3.3.2.2.A	One feedwater system and main turbine high water level trip channel inoperable.	Reactor high level Feedwater system and main turbine trip instrumentation.	Yes	Feedwater system and main turbine trip to prevent water intrusion into turbines	Two of three channels	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.3.4.1.A	End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation - One or more required channels inoperable.	Recirculation Pump Trip (see Note 2)	Not explicitly	Recirculation Pump Trip	One of two channels	Same	Failure of recirculation pump breakers to trip will be used as a surrogate for the RICT calculation. The surrogate is conservative as the breakers are tripped by the EOC-RPT logic. Thus, the RICT calculated for this surrogate is bounding for each channel because one channel out of service does not prevent the trip of the RPT breakers. Per TSTF-505, Rev 2, Table 1 Note 1, RICT is not applicable when all required channels are inoperable.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.3.4.2.A	Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation - One or more channels inoperable.	The ATWS-RPT System includes sensors, relays, bypass capability, circuit breakers, and switches that are necessary to cause initiation of a recirculation pump trip (see Note 3)	Not explicitly	Recirculation Pump Trip	One of two trains	Same	Failure of recirculation pump breakers will be used as a surrogate in the RICT calculation.
3.3.5.1.B	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	ECCS actuation instrumentation for low pressure core spray (LPCS), low pressure coolant injection (LPCI) and high pressure core spray (HPCS) (See Notes 4, 5)	Yes	Initiate ECCS (HPCS, LPCS, and LPCI)	One of two channels for LPCI, LPCS, one of four channels for HPCS	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.3.5.1.C	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	See 3.3.5.1.B					
3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	The ECCS instrumentation actuating HPCS (See Notes 4, 5)	Yes	Initiate HPCS	One of two channels	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.3.5.1.E	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	ECCS actuation instrumentation for LPCS, LPCI, HPCS (See Notes 4, 5)	Yes	Initiate ECCS (HPCS, LPCS, and LPCI)	One channel per system	Same	HPCS discharge instrumentation is not modeled, Failure of the HPCS min flow valve will be used as a surrogate for HPCS discharge instrumentation failure
3.3.5.1.F	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	ADS initiation logic and instrumentation	Not explicitly	Initiate ADS	One of two channels	Same	Failure of all ADS valves to open will be used as a conservative surrogate for the RICT calculation.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.3.5.1.G	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	See 3.3.5.1.F					
3.3.5.3.B	RCIC System Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	Reactor Vessel Level Instrumentation supporting RCIC	Yes	RCIC initiation	Two of four channels, two channels per division	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.
3.3.5.3.D	RCIC System Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	RCIC Pump Suction Pressure Instrumentation	Not explicitly	Swap suction source from the CST to the Suppression Pool	One of two channels	Same	Failure to swap suction sources will lead to failure of the pump. RCIC suction from the CST is directly modeled in the PRA and failure of this function will be used as a bounding surrogate in the RICT calculations

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.3.6.1.A	Primary Containment Isolation Instrumentation - One or more channels inoperable.	Sensors, relays and switches that are necessary to cause initiation	Yes	Automatic isolation of Primary Containment Isolation valves	Logic strings based on subsystem logic structure	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.
3.3.7.2.A	Mechanical Vacuum Pump Isolation Instrumentation - One or more channels inoperable	Detectors, monitors, and relays to trip and isolate the mechanical vacuum pumps.	No	Trip and isolate the mechanical vacuum pumps	One of two channels in each of two trip systems	N/A	SSCs are not modeled. The model will be updated to include these SSCs prior to exercising the RICT program for this TS. The PRA Success Criteria will match the Design Success Criteria. Failure to mitigate ISLOCA/HELB initiating events has been used as a surrogate representation of the risk for the Table E1-2 sample calculations.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.3.8.1.A	Loss of Power (LOP) Instrumentation – One or more required channels inoperable.	The LOP System includes sensors, relays, bypass capability, circuit breakers, and switches that are necessary to trip offsite power circuits and start the emergency diesel generators. (See Note 6)	Not explicitly	Trip offsite power circuits.	Two of three channels	Same	Failure of the EDG output breakers and divisional switchgear will be used as a conservative surrogate for the RICT calculation. The surrogate is conservative as the breakers are actuated by the LOP system to trip offsite power and start the EDGs. Thus, the RICT calculated for this surrogate is bounding for each channel. Per TSTF-505, Rev 2, Table 1 Note 1, RICT is not applicable when all required channels are inoperable.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.5.1.A	One low pressure ECCS injection/spray subsystem inoperable.	Three LPCI trains and one LPCS train	Yes	Low pressure injection into the RPV	Two of four subsystems	One of four subsystems	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis for each train.
3.5.1.B	High Pressure Core Spray (HPCS) System inoperable.	One train of HPCS	Yes	High pressure injection into the RPV	One of one trains	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.5.1.C	Two ECCS injection subsystems inoperable. OR One ECCS injection and one ECCS spray subsystem inoperable.	One train of HPCS, one train of LPCS, and three trains of RHR	Yes	RPV inventory control and decay heat removal	Two of five injection subsystems and one of two spray subsystems	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.
3.5.1.E	One required ADS valve inoperable.	Seven ADS valves and supporting components	Yes	RPV rapid depressurization	Five of seven ADS valves	Five of seven ADS valves in ATWS scenarios. Two of seven ADS valves in non-ATWS scenarios.	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.5.1.F	One required ADS valve inoperable AND One low pressure ECCS injection/spray subsystem inoperable.	See 3.5.1.A and 3.5.1.E					
3.5.3.A.2	RCIC System Inoperable.	One train of RCIC	Yes	Supply high pressure makeup water to the RPV.	One of one trains	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.
3.6.1.2.C.3	One or more primary containment air locks inoperable for reasons other than Condition A or B.	Primary Containment Air Locks	No	Isolate primary containment during personnel entry and exit.	One of two containment air lock doors closed.	N/A	See Note 8 The airlocks are not modeled so a large pre-existing leak failure will be used as a conservative surrogate for the RICT calculation.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.6.1.3.A	One or more penetration flow paths with one PCIV inoperable except due to leakage not within limit.	Primary Containment Isolation Valves (PCIVs)	Yes	Limit fission product release during and following postulated accidents	One of two isolation valves per penetration	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. Any PCIV not explicitly modeled will use a pre-existing small leak event for the RICT calculation. The success criteria are consistent with the design basis.
3.6.1.3.E	One or more penetration flow paths with one or more containment purge exhaust valves not within purge valve leakage limits.	Containment purge exhaust valves	Yes	Limit fission product release during and following postulated accidents	One of two isolation valves per penetration deactivated or locked closed or administratively verified closed	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.6.1.6.A	One RHR drywell spray subsystem inoperable.	Two trains of RHR (See Note 7)	Yes	Removal of heat from the drywell and pressure suppression	One of two trains	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.
3.6.1.7.A	One line with one or more suppression chamber-to-drywell vacuum breakers inoperable for opening.	Four lines with two vacuum breakers in series per line	Yes - partially modeled	Relieve vacuum in the drywell	Three of four lines	Same	The PRA model includes one failure mode: lines fail to close after initially opening. The model will be updated to include this failure mode prior to exercising the RICT program for this TS. The PRA Success Criteria will match the Design Success Criteria.
3.6.2.3.A	One RHR suppression pool cooling subsystem inoperable.	Two trains of RHR (See Note 7)	Yes	Removal of heat from the Suppression Pool for DBA LOCA and transient events	One of two trains	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.6.2.4.A	One RHR suppression pool spray subsystem inoperable.	Two trains of RHR (See Note 7)	Yes	Removal of heat from the Suppression Pool and suppression pool airspace	One of two trains	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.
3.7.1.A	One SW supply header cross connect valve inoperable.	SW cross-connect valves	Yes	Supply cooling water to safety related components	One of two crossties	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.7.1.C	One SW subsystem inoperable for reasons other than Conditions A and B.	Two SW trains each with three pumps	Yes	Supply cooling water to safety related components	Two of six pumps during a LOCA concurrent with a LOOP. Three of six pumps during a LOCA without a LOOP and UHS temperature <= 82°F. Four of six pumps during a LOCA without a LOOP and UHS temperature > 82°F and <= 84°F.	Same, except UHS temperature is assumed to be <= 82°F	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis except when UHS temperature is > 82°F. The model is being updated to include this condition prior to exercising the RICT program for this TS.
3.7.1.D	One division of intake deicer heaters inoperable.	Eighty-four intake deicer heaters across two divisions in each intake structure	Not explicitly	Ensures the necessary intake structure suction area is available	Fourteen heaters in each intake structure across one division.	Same	The intake deicer heaters are not directly modeled in the PRA. The model will be updated to explicitly include these components prior to its use with RICT for this TS.
3.7.1.E	One required SW pump not in operation.	See 3.7.1.C					

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.7.1.F	Two or more required SW pumps not in operation.	See 3.7.1.C					
3.7.5.A	Main Turbine Bypass System - Requirements of the LCO not met.	Turbine Bypass Valves	Yes	Control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown.	Five of five bypass valves	Three of five bypass valves	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. PRA success criteria is based on the minimum valves required to prevent major demands on the suppression pool.
3.8.1.A	One required offsite circuit inoperable.	Lines 5/6	Yes	Supply AC loads during operation	One offsite source	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.8.1.B	One required DG inoperable.	2 EDGs and HPCS EDG	Yes	Supply AC loads when offsite power is lost	One of two non HPCS EDGs .	Same, except HPCS EDG alignment to either Div I or Div II switchgear is credited.	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.
3.8.1.C	Two required offsite circuits inoperable.	See 3.8.1.A					
3.8.1.D	One required offsite circuit inoperable AND One required DG inoperable.	See 3.8.1.A and 3.8.1.B					
3.8.4.A	Division 1 or 2 DC electrical power subsystem inoperable.	Two Battery chargers and one battery per division	Yes	Supply DC loads during operation	One of two subsystems	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.8.7.A	One emergency UPS inverter inoperable.	Two UPS inverters per division	Yes	AC power conditioning for the required divisional loads	One of two UPS inverters per division	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis.
3.8.8.A	One or both Division 1 and 2 AC electrical power distribution subsystems inoperable.	4.16kV buses, 600V load centers and distribution panels, and 120V panels	Yes	AC power distribution to the required divisional loads	One of two divisions of distribution	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis. The RICT program will not be entered if both divisions are inoperable as this is a loss of function.

List of Revised Required Actions to Corresponding PRA Functions

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

Technical Specification (TS)	TS Condition Description	SSCs Covered by TS LCO Condition	SSCs Modeled in PRA?	Function Required by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Other Comments
3.8.8.B	One or both Division 1 and 2 120 VAC uninterruptible electrical power distribution subsystems inoperable.	Two UPS per division	Yes	AC power distribution to the required divisional loads	One of four UPSs	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis. The RICT program will not be entered if both divisions are inoperable as this is a loss of function.
3.8.8.C	One or both Division 1 and 2 DC electrical power distribution subsystems inoperable.	Two divisions of DC distribution	Yes	DC power distribution to the required divisional loads	One of two divisions	Same	SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. The success criteria are consistent with the design basis. The RICT program will not be entered if both divisions are inoperable as this is a loss of function.

List of Revised Required Actions to Corresponding PRA Functions

1. The reactor protection system is made up of two independent trip systems (A and B). Each trip system contains 2 logic channels (A1, A2 and B1, B2). The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. Each channel contains the various functional inputs to RPS such Reactor level, MSIV closure, etc. Loss of any functional input does not prevent the channel from responding to other inputs. Use of an electrical SCRAM failure as a surrogate for a non-modeled functional input is conservative as it encompasses loss of all the inputs to all channels rather than any single input to a channel.

2. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

3. ATWS-RPT system instrumentation is part of the redundant reactivity control system and has 2 independent trip systems each composed of two channels of each functional input. Each trip system uses a 2-out-of-2 logic for each function. Thus, either two Reactor Water Level – Low Low, Level 2 or two Reactor Vessel Steam Dome Pressure – High signals are needed to trip a trip system. Either trip system will trip both recirculation pump fast speed breakers. A trip on low RPV level will also trip the low frequency motor generator (LMFG) breakers. A trip on high RPV pressure will send a start signal to the LFMGs. The LFMGs are then tripped after 25 seconds if RPV pressure signals are still high and the associated APRM signals are not downscale (approximately ≤ 4% RTP).

4. The control logic for LPCS automatic initiation occurs for low reactor water level or high drywell pressure. Reactor Vessel water level – Low Low Low (Level 1) is sensed by two trip units in the Reactor Instrumentation System. Drywell pressure – High signals are sent from the Reactor Instrumentation System (RIS) to two high drywell pressure relay contacts. The trip unit outputs are in a two-out-of-two logic. The outputs of the four trip units (two trip units from each of the two variables) are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for automatic initiation. Automatic initiation of LPCI occurs for conditions of Reactor Vessel Water Level – Low Low Low, Level 1 or Drywell Pressure – High. Reactor vessel water level is monitored by two redundant differential pressure transmitters per division and drywell pressure is monitored by two redundant pressure transmitters per division, each providing input to a trip unit. The outputs of the four Division 2 LPCI (loops B and C) trip units (two trip units from each of the two variables) are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic. The Division 1 LPCI (loop A) receives its initiation signal from the LPCS logic, which uses a similar one-out-of-two taken twice logic. Automatic initiation of HPCS occurs for conditions of Reactor Vessel Water Level – Low Low, Level 2 or Drywell Pressure – High. Reactor vessel water level is monitored by four redundant differential pressure transmitters and drywell pressure is monitored by four redundant pressure transmitters, each providing input to a trip unit. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each variable.

List of Revised Required Actions to Corresponding PRA Functions

5. Individual pieces of instrumentation such as a pressure transmitter may be shared by multiple design basis functions.

6. The LOP instrumentation consists of two different undervoltage functions: loss of voltage and degraded voltage. Each function is monitored by three separate undervoltage relays, one relay per phase. The relay outputs are arranged in a two-out-of-three logic configuration for each division. The relay outputs supply input to the time delay functions for each undervoltage function. All divisions have one time delay relay for loss of voltage. Division 1 and 2 utilize two time delay relays for degraded voltage. Division 3 has only one time delay relay for degraded voltage. When a loss of voltage or degraded voltage setpoint has been exceeded and the respective time delay completed, the time delay relay actuates and sends a LOP signal to the respective bus load shedding control scheme, which starts the associated diesel generator (DG), provides a closure signal for the DG output breaker, opens both offsite circuit supply breakers, and for Division 1 and 2 only, sheds all loads on the 4.16kV emergency bus, including the stub bus (except the 600 V load centers).
.
7. The RHR system contains three separate pump trains, two of which contain heat exchangers for heat removal. The third pump train is for LPCI functions only.

8. As indicated in Table E1-1, the containment air locks are not explicitly modeled in the NMP2 PRA. Since the containment airlocks are not modeled, there are no explicit PRA Success Criteria. However, this condition will be modeled as early containment bypass as a conservative surrogate in the PRA. Compliance with the remaining portions of LCO Condition 3.6.1.2 ensure that there is a physical barrier (i.e., closed door) and an acceptable overall leakage from containment. Thus, the function is still maintained. Required Action C.1 of LCO Condition 3.6.2 requires the condition to be assessed in accordance with TS 3.6.1, "Containment Integrity" (i.e., "Initiate action to evaluate overall containment leakage rate per LCO 3.6.1" with a Completion Time of Immediately.)

List of Revised Required Actions to Corresponding PRA Functions

Table E1-2: Example RICT Calculations		
Tech Spec	**TS Condition**	**RICT Estimate[1,2]**
3.1.7.A	One SLC subsystem inoperable.	720
3.3.1.1.A	RPS Instrumentation - One or more required channels inoperable.	500
3.3.1.1.B	RPS Instrumentation - One or more Functions with one or more required channels inoperable in both trip systems.	500
3.3.2.2.A	One feedwater system and main turbine high water level trip channel inoperable.	720
3.3.4.1.A	End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation - One or more required channels inoperable.	720
3.3.4.2.A	Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation - One or more channels inoperable.	720
3.3.5.1.B	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	28
3.3.5.1.C	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	248
3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	720
3.3.5.1.E	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	248
3.3.5.1.F	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	106
3.3.5.1.G	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	106
3.3.5.3.B	RCIC System Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	720
3.3.5.3.D	RCIC System Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.3-1.	720
3.3.6.1.A	Primary Containment Isolation Instrumentation - One or more channels inoperable	720
3.3.7.2.A	Mechanical Vacuum Pump Isolation Instrumentation - One or more channels inoperable	98
3.3.8.1.A	Loss of Power (LOP) Instrumentation - One or more required channels inoperable.	253
3.5.1.A	One low pressure ECCS injection/spray subsystem inoperable.	720
3.5.1.B	High Pressure Core Spray (HPCS) System inoperable.	720
3.5.1.C	Two ECCS injection subsystems inoperable OR One ECCS injection and one ECCS spray subsystem inoperable.	720
3.5.1.E	One required ADS valve inoperable.	720
3.5.1.F	One required ADS valve inoperable AND One low pressure ECCS injection/spray subsystem inoperable.	720
3.5.3.A.2	RCIC System Inoperable.	720
3.6.1.2.C.3	One or more primary containment air locks inoperable for reasons other than Condition A or B.	720

List of Revised Required Actions to Corresponding PRA Functions

Table E1-2: Example RICT Calculations

Tech Spec	TS Condition	RICT Estimate[1,2]
3.6.1.3.A	One or more penetration flow paths with one PCIV inoperable except due to leakage not within limit.	720
3.6.1.3.E	One or more penetration flow paths with one or more containment purge exhaust valves not within purge valve leakage limits.	720
3.6.1.6.A	One RHR drywell spray subsystem inoperable.	415
3.6.1.7.A	One line with one or more suppression chamber-to-drywell vacuum breakers inoperable for opening.	366
3.6.2.3.A	One RHR suppression pool cooling subsystem inoperable.	415
3.6.2.4.A	One RHR suppression pool spray subsystem inoperable.	415
3.7.1.A	One SW supply header cross connect valve inoperable.	645
3.7.1.C	One SW subsystem inoperable for reasons other than Conditions A and B.	139
3.7.1.D	One division of intake deicer heaters inoperable.	17
3.7.1.E	One required SW pump not in operation.	463
3.7.1.F	Two or more required SW pumps not in operation.	139
3.7.5.A	Main Turbine Bypass System - Requirements of the LCO not met.	720
3.8.1.A	One required offsite circuit inoperable.	101
3.8.1.B	One required DG inoperable.	524[4]
3.8.1.C	Two required offsite circuits inoperable.	145
3.8.1.D	One required offsite circuit inoperable AND One required DG inoperable.	17
3.8.4.A	Division 1 or 2 DC electrical power subsystem inoperable.	720[3]
3.8.7.A	One emergency UPS inverter inoperable.	720
3.8.8.A	One or both Division 1 and 2 AC electrical power distribution subsystems inoperable.	22[3]
3.8.8.B	One or both Division 1 and 2 120 VAC uninterruptible electrical power distribution subsystems inoperable.	28[3]
3.8.8.C	One or both Division 1 and 2 DC electrical power distribution subsystems inoperable.	76[3]

1. RICTs are based on the internal events, internal flood, and internal fire PRA model calculations with seismic penalties. RICTs calculated to be greater than 30 days are capped at 30 days based on NEI 06-09, Revision 0-A. RICTs are rounded to the nearest number of hours for illustrative purposes.

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

3. Based on a loss of one division of electric power only.

4. Div I/II Emergency Diesel Generators

List of Revised Required Actions to Corresponding PRA Functions

Table E1-3 lists the TSTF-505 Rev 2 Table 1 Tech Specs that require additional justification along with a description of how the additional justification is provided in the LAR.

Table E1-3: TSTF-505 Rev 2 Table 1 Tech Specs that Require Additional Justification

Tech Spec Description	TSTF-505 Tech Spec	NMP2 Tech Spec	Additional Justification
Source Range Monitor Instrumentation - One or more required SRMs inoperable in MODE 2 with intermediate range monitors (IRMs) on Range 2 or below.	3.3.1.2.A	3.3.1.2.A	N/A – TSTF-505 changes are excluded.
Feedwater and Main Turbine High Water Level Trip Instrumentation - Two or more feedwater and main turbine high water level trip channels inoperable.	3.3.2.B	3.3.2.B.1	N/A – TSTF-505 changes are excluded.
End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation - One or more required channels inoperable.	3.3.4.1.A.1 3.3.4.1.A.2	3.3.4.1.A.1 3.3.4.1.A.2	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained. Failure of recirculation pump breakers to trip will be used as a surrogate for the RICT calculation. The surrogate is conservative as the breakers are tripped by the EOC-RPT logic. Thus, the RICT calculated for this surrogate is bounding for each channel because one channel out of service does not prevent the trip of the RPT breakers.
Low-Low-Set (LLS) Instrumentation	3.3.6.3	----	N/A - The NMP2 TS do not contain this TS. Therefore, a change is not proposed to the NMP2 TS.
Loss of Power (LOP) Instrumentation - One or more channels inoperable.	3.3.8.1.A	3.3.8.1.A.1	TSTF-505 changes are incorporated. However, under certain circumstances, with more

List of Revised Required Actions to Corresponding PRA Functions

Table E1-3: TSTF-505 Rev 2 Table 1 Tech Specs that Require Additional Justification

Tech Spec Description	TSTF-505 Tech Spec	NMP2 Tech Spec	Additional Justification
			than one channel inoperable, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when trip capability is not maintained.
Primary Containment Air Lock - Primary containment air lock inoperable for reasons other than Condition A or B.	3.6.1.2.C.3	3.6.1.2.C.3	TSTF-505 changes are incorporated. However, under certain circumstances, with more than one primary containment airlocks inoperable, excessive leakage or a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when leakage exceeds limits or it there is a loss of function.
Primary Containment Isolation Valves (PCIVs) - One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits.	3.6.1.3.E.1	3.6.1.3.E.1	Note: NMP2 TS 3.6.1.3.E.1 specifies containment purge exhaust valves. TSTF-505 changes are incorporated. However, under certain circumstances, with more than one containment purge valve not within leakage limits, a loss of function may occur. Therefore, a Note is added to the Completion Time which prohibits applying a RICT when there is a loss of function.
Reactor Building-to-Suppression Chamber Vacuum Breakers	3.6.1.7	----	N/A - The NMP2 TS do not contain this TS. Therefore, a change is not proposed to the NMP2 TS.
Main Turbine Bypass System - Requirements of the LCO not met or Main Turbine Bypass System inoperable.	3.7.7.A	3.7.5.A.1	TSTF-505 changes are incorporated. SSCs are modeled consistent with the TS scope and so can be directly included in the RTR tool for the RICT program. PRA success criteria is based on calculations for the minimum valves required to prevent major demands on the suppression pool.

List of Revised Required Actions to Corresponding PRA Functions

Table E1-3: TSTF-505 Rev 2 Table 1 Tech Specs that Require Additional Justification			
Tech Spec Description	**TSTF-505 Tech Spec**	**NMP2 Tech Spec**	**Additional Justification**
			The TS function to limit peak pressure in the main steam lines and to maintain reactor pressure within acceptable limits during events that cause rapid pressurization is the PRA modeled function. The combined pressure control function of the turbine control valves and bypass valves while the main turbine is online is not modeled but this is not a mitigation function that would affect risk.

2. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (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. 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).

ENCLOSURE 2

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

ENCLOSURE 2

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

1. Introduction

This enclosure provides information on the technical adequacy of the Nine Mile Point Unit 2 (NMP2) Probabilistic Risk Assessment (PRA) internal events model (including flooding) and the NMP2 fire PRA model in support of the license amendment request to revise Technical Specifications to implement NEI 06-09, Revision 0-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines” (Reference 10).

Topical Report NEI 06-09, Revision 0-A, as clarified by the NRC final safety evaluation of this report (Reference 11), defines the technical attributes of a PRA model and its associated Configuration Risk Management Program (CRMP) tool, presently referred to as the Real Time Risk (RTR) tool, required to implement this risk-informed application. Meeting these requirements satisfies Regulatory Guide (RG) 1.174 (Reference 12) requirements for risk-informed plant-specific changes to a plant's licensing basis.

Exelon employs a multi-faceted approach to establishing and maintaining the technical acceptability and fidelity of PRA models for all operating Exelon Nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process and the use of self-assessments and independent peer reviews.

Section 2 outlines requirements related to the scope of the NMP2 PRA Full Power Internal Events (FPIE) model including internal flooding. Section 3 outlines requirements related to the scope of the NMP2 Fire PRA (FPRA) model. Section 4 describes the PRA models used in the respective license amendment applications. Section 5 describes the disposition and resolution of PRA peer review findings and self-assessment items.

2. Requirements Related to Scope of NMP2 PRA Models

The PRA models discussed in this enclosure have been assessed against RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 2 (Reference 1) consistent with NRC RIS 2007-06 (Reference 2).

Finding and Observation (F&O) closure reviews were conducted on the PRA models discussed in this enclosure. Closed findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, “Close-out of Facts and Observations” (F&Os) (Reference 3) as accepted by NRC in the letter dated May 3, 2017 (Reference 4).

Note that this portion of the NMP2 PRA model does not incorporate the risk impacts of external events. The treatment of seismic risk and other external hazards for this application are discussed in Enclosure 4.

3. Scope and Technical Adequacy of NMP2 Internal Events and Internal Flooding PRA Model

The NMP2 FPIE PRA model was peer reviewed in July 2009 using the NEI 05-04 process, the PRA Standard (ASME/ANS RA-Sa-2009)(Reference 9) and Regulatory Guide 1.200, Revision 2. This Peer Review (Reference 5) was a full-scope review of the technical elements of the

internal events and internal flooding, at-power PRA. The findings from the peer review have been addressed in the internal events PRA model.

In February 2019, an F&O Closure Review was conducted for NMP2. The scope of the review included the internal events and internal flooding peer review findings. The F&O Independent Assessment Team determined that there was one upgrade to the model, (addition of support system initiating event fault trees) which necessitated a focused-scope peer review. The focused-scope peer review determined there are three Finding level F&Os resulting in four Not-Met SRs. Each finding has been dispositioned in Table E2-1. All other findings were closed during the February 2019 F&O Closure Review (Reference 6).

Table E2-1 provides the open and partially resolved findings from the F&O closure review. The current status of the findings and the potential impact of these open findings on the 4b application is also noted. For all entries, the "Impact to TSTF-505 Implementation" column reflect the assessment of the finding closure review, with an assessment relative to the respective application added for all open or partially resolved items.

Given the impact on RICT calculations from F&Os will be minor, the NMP2 internal events PRA will be of adequate technical capability to support the TSTF-505 program.

4. Scope and Technical Adequacy of NMP2 Fire PRA Model

The NMP2 Fire PRA (FPRA) peer review (Reference 7) was performed in June 2018 using the NEI 07-12 Fire PRA peer review process (Reference 8), the ASME PRA Standard, ASME/ANS RA-Sa-2009 (Reference 9) and Regulatory Guide 1.200, Rev. 2 (Reference 1). The purpose of this review was to establish the technical acceptability of the FPRA for the spectrum of potential risk-informed plant licensing applications for which the FPRA may be used. The FPRA peer review was a full-scope review of all of the technical elements of the NMP2 at-power FPRA against all technical elements in Part 4 of the ASME/ANS PRA Standard, including the referenced internal events supporting requirements (SRs) in Part 2[A1][A2].

The findings from the Fire RPA peer review have been addressed in the Fire PRA model. In February 2019, an F&O Closure Review was conducted for NMP2 (Reference 6). The scope of the review included fire peer review findings. All of the findings from the 2018 fire PRA peer review were resolved. Currently, there are no open findings against the fire PRA model (Reference 6).

Given there are no partially resolved or open findings that may impact RICT calculations, the NMP2 PRA is of adequate technical capability to support the TSTF-505 program.

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

TABLE E2-1
NMP2 FOCUSED-SCOPE PEER REVIEW OPEN FINDINGS IMPACT ON RICT SUBMITTAL

Finding ID	Originating SR	Finding	Status	Disposition	Impact to TSTF-505 Implementation
5-1	IE-C13	A correction factor was used with each SSIE fault tree generated initiating event frequency. These factors are applied in the fault tree logic. The purpose of the correction factor is to allow the fault tree cutsets to propagate through the model and still produce results similar to the baseline (quantification with the point estimates for the SSIEs). The purpose of the support systems fault trees is to replace the point estimate values (Bayesian update of generic data with plant experience) with plant-specific system models to better reflect the actual plant behavior. This incorporates plant-specific characteristics into the model to both provide a better plan specific estimation of the initiating event frequencies and to gain insights into the causes of these initiating events from the importance values of the components in the models. The use of these correction factors negates the benefits of using fault trees for SSIEs. These factors defeat the purpose of incorporation of plant specific characteristics in terms of the relative importance of the SSIEs and skew the importance of contributing SSCs to overall CDF. The factors used range from ~0.67 to ~5.9E-4 and have a considerable impact on the results. No justification is provided for these correction factors or their applicability to the model and there is no discussion of the differences between the SSIE fault tree frequencies and the point estimate frequencies.	Open	Support System Initiating event fault trees can produce non-representative results when 24-hour-based basic events are adjusted to apply a year-long mission time. NMP2 employed correction factors to assure results were representative of plant experience. This finding has a small impact on PRA model results.	Improved modeling without correction factors will be incorporated into the internal events PRA model prior to implementation of the RICT program, so there will be no impact on RICT calculations.

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

TABLE E2-1
NMP2 FOCUSED-SCOPE PEER REVIEW OPEN FINDINGS IMPACT ON RICT SUBMITTAL

Finding ID	Originating SR	Finding	Status	Disposition	Impact to TSTF-505 Implementation
8-1	IE-C3, HR-H1	There is no indication that a systematic review of the cut sets for each SSIE was conducted to identify the potential for recovery actions that could have prevented an upset in one of the modeled systems from proceeding to an actual initiating event. This is expected as part of ensuring that the frequencies are calculated in as realistic a manner as possible.	Open	NMP2 has employed correction factors to assure results are representative of plant experience; implicitly crediting recovery. This finding has a small impact on PRA model results.	A systematic review of the cutsets comprising the SSIEs to identify any potential recovery actions and an assessment for any actions that are feasible and that would affect the frequencies of associated SSIEs will be performed prior to implementation of the RICT program, so there will be no impact on RICT calculations.
8-2	IE-C10	Some potentially important common-cause contributors are not properly assessed for initiating event SWPX. The events do not appropriately characterize failure to run over the course of a year. The common-cause events that include failures of normally operating components should be adjusted to include the failure to run per year.	Open	Support System Initiating event fault trees can produce non-representative results when 24-hour-based basic events are adjusted to apply a year-long mission time. NMP2 used a 24 hour CCF mission time to avoid dominant contribution from CCF events which have not been prevalent in plant or industry Initiating Event experience. This finding has a small impact on PRA model results. .	This simplification will be removed and appropriate CCF mission times mission will be used prior to implementation of the RICT program. Therefore, there will be no impact on RICT calculations.

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

5. References

1. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, US Nuclear Regulatory Commission, March 2009.

2. NRC Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," March 22, 2007.

3. NEI Letter to NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)," February 21, 2017, Accession Number ML17086A431.

4. NRC Letter to Mr. Greg Krueger (NEI), "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 7-12, and 12-13, Close Out of Facts and Observations (F&Os)," May 3, 2017, Accession Number ML17079A427.

5. "Nine Mile Point Unit 2 PRA Peer Review Report Using ASME PRA Standard Requirements, March 2010.

6. Jensen Hughes Report 032405-RPT-01, "Nine Mile Point Unit 2 PRA Fact and Observation Independent Assessment & Focused Scope Peer Review," April 2019.

7. “Nine Mile Point Nuclear Station Unit 2 Fire PRA Peer Review Report Using ASME/ANS PRA Standard Requirements, October 2018.

8. NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Revision 1, June 2010.

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

10. 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, October 12, 2012 (ADAMS Accession No. ML12286A322).

11. Letter from Jennifer M. Golder (NRC) to Biff Bradley (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).

Information Supporting Consistency with Regulatory Guide 1.200, Revision 2

12. 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 2, May 2011.

ENCLOSURE 3

License Amendment Request

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Docket No. 50-410

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

Information Supporting Technical Adequacy of PRA Models Without PRA Standards Endorsed by Regulatory Guide 1.200, Revision 2

This enclosure is not applicable to the NMP2 submittal. Exelon is not proposing to use any PRA models in the NMP2 Risk-Informed Completion Time Program for which a PRA standard, endorsed by the NRC in RG 1.200, Revision 2 does not exist.

ENCLOSURE 4

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

Topical Report NEI 06-09, Revision 0-A (Reference 1), as clarified by the Nuclear Regulatory Commission (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 as well as discuss conservative or bounding 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. This enclosure also provides the Nine Mile Point 2 (NMP2) specific results of the application of the generic methodology and the disposition of impacts on the NMP2 Risk Informed Completion Time (RICT) Programs. Section 3 of this enclosure presents the plant-specific analysis of seismic risk to NMP2. Section 4 of this enclosure presents the justification for excluding analysis of high wind risk to NMP2. Section 5 presents the justification for excluding External Flooding for NMP2. Section 6 of this enclosure presents the justification for excluding analyses of other external hazards from the NMP2 PRA.

Topical Report NEI 06-09 does not provide a specific list of hazards to be considered in a RICT Program. However, non-mandatory Appendix 6-A in the ASME/ANS PRA Standard (Reference 3) provides a guide for identification of most of the possible external events for a plant site. Additionally, NUREG-1855 (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. This information was reviewed for the NMP2 site and augmented with a review of information on the site region and plant design to identify the set of external events to be considered. The information in the USAR regarding the geologic, seismologic, hydrologic, and meteorological characteristics of the site region as well as present and projected industrial activities in the vicinity of the plant were also reviewed for this purpose. No new site-specific and plant-unique external hazards were identified through this review. The list of hazards in Appendix 6-A of the PRA Standard were considered for NMP2 as summarized in Table E4-8.

The scope of this enclosure is consideration of the hazards in Table E4-8 for NMP2. As explained in subsequent sections of this enclosure, risk contribution from seismic events is evaluated quantitatively, and the other listed external hazards are evaluated and screened as having low risk.

2 Technical Approach

The guidance contained in NEI 06-09 states that all hazards that contribute significantly to incremental risk of a configuration must be quantitatively addressed 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 to be part of the RICT Program.

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

Consistent with NUREG-1855, external hazards may be addressed by:

1) Screening the hazard based on a low frequency of occurrence,
2) Bounding 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, seismic events greater than the design-basis earthquake (DBE), etc. These beyond design basis conditions challenge the capability of the 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 or seismic events causing loss of offsite power, etc. While the plant design basis assures that the safety related equipment necessary to respond to these challenges are protected, the occurrence of these conditions nevertheless causes a demand on these systems that presents a risk.

Hazard Screening

The first step in the evaluation of an external hazard is screening based on an estimation of a bounding core damage frequency (CDF) for beyond design basis hazard conditions. An example of this type of screening is reliance on the NRC's 1975 Standard Review Plan (SRP) (Reference 5), which is acknowledged in the NRC's Individual Plant Examination of External Events (IPEEE) procedural guidance (Reference 6) as assuring a bounding CDF of less than 1E-6/yr for each hazard. The bounding 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 bounding conditional core damage probability (CCDP) for those conditions. If the bounding CDF for the hazard can be shown to be less than 1E-6/yr, then beyond design basis challenges from that hazard can be screened out and do not need to be addressed 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 this RICT is 30 days.

- If the maximum CDF contribution from a hazard is <1E-6/yr, then the maximum ICDP from the hazard is <1E-7 (1E-6/yr * 30 days/365 days/yr).

- Thus, the bounding ICDP contribution from the hazard is shown to be less than 1% of the permissible ICDP in the bounding time for the condition. Such a minimal contribution is not significant to the decision in computing a RICT.

The NMP2 IPEEE hazard screening analysis (Reference 7) has been updated to reflect current NMP2 site conditions as shown in subsequent sections. The results show that all the events listed in Table E4-8 can be screened except seismic events for NMP2.

While the direct CDF contribution from beyond design basis hazard conditions can be shown to be non-significant 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 6.

Hazard Analysis - CDF

There are two options in cases where the bounding CDF for the external hazard cannot be shown to be less than 1E-6/yr. 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 bounding CDF contribution for the hazard.

Evaluate Bounding 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 bounding conditional large early release probability (CLERP) that aligns with the bounding CDF evaluation. The incremental large early release frequency (ILERF) is then computed as follows:

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

The approach used for seismic LERF is described in Section 3.

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

Risks from Hazard Challenges

Given the selection of an estimated bounding CDF/LERF, the approach considered must assure that the RICT Program calculations reflect the change in CDF/LERF caused by the out of service equipment. For NMP2, as discussed later in this enclosure, the only beyond design basis hazard that could not be screened out is the seismic hazard, and the approach used considers that the change in risk with equipment out of service will not be higher than the bounding seismic CDF.

The above steps address the direct risks from damage to the facility from external hazards. While the direct CDF contribution from beyond design basis hazard conditions can be shown to be non-significant using these steps without a full PRA, there are risks that may be addressed. These risks 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 the design basis levels can cause extended loss of offsite power conditions. Additionally, depending on the site, external floods can challenge the availability of normal plant heat removal mechanisms.

The approach 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 risk.

Section 3 of this enclosure provides the analysis for the NMP2 site with respect to the beyond design basis seismic hazard, and Section 4 provides an analysis for the extreme winds hazard. Section 5 address the analysis of External Flooding for NMP2. Section 6 of this enclosure provides an analysis of the representative external hazards for the NMP2 site.

3 Seismic Analysis

The Seismic Margin Assessment (SMA) was performed using the EPRI SMA method (Reference 8) and a seismic PRA was developed to the guidelines contained in Generic Letter 88-20 Supplement 4 and NUREG 1407. The review level earthquake (RLE) used for screening was chosen as 0.5g, rather than 0.3g, as recommended by NUREG-1407. The NMP2 site exhibits very low seismicity, but a conservative Regulatory Guide 1.60 (Reference 9) safe shutdown earthquake (SSE) with a 0.15g peak ground acceleration was adopted as a design basis. The Operating Basis Earthquake (OBE) is herein defined as an earthquake that would produce a horizontal acceleration at the site of 0.075 g (Reference 10).

Since the seismic PRA (SPRA) developed for the NMP2 IPEEE was not maintained, an alternative approach was taken to provide an estimate of seismic core damage frequency (SCDF) based on convolving the current NMP2 seismic hazard curve (Reference 11), with the NMP2 IPEEE-determined plant HCLPF, whose seismic failure would lead directly to core damage.

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

The calculation of seismic large early release frequency (SLERF) is performed by estimating an average seismic conditional large early release probability (SCLERP), based on the spectrum of SCDF accident sequence types, and multiplying SCDF by the averaged SCLERP estimate.

Inputs and Assumptions

1. Hazard Curve: The NMP2 seismic hazard is defined by the seismic hazard curve (SHC) provided to NRC in Reference 11.

2. PGA Metric: The ground motion metric used to define the seismic hazard in this analysis is peak ground acceleration (PGA). The PGA curve in Reference 11 was compared with other frequencies (i.e., other spectral hazard frequencies) for ground motion. A sensitivity analysis was performed in Reference 12 that uses the 2.5 Hz hazard curve (i.e., industry SPRAs that do not use PGA instead often use a hazard curve in the 2-5 Hz range) because it intersects with the PGA hazard curve. Another sensitivity analysis was performed in Reference 12 to evaluate the effect on the calculated risk results by changing the seismic hazard intervals used in the convolution.

3. High Confidence Low Probability of Failure (HCLPF): The NMP2 plant high confidence low probability of failure (HCLPF), as determined by the seismic margin assessment (SMA), is greater than 0.5g. With one exception, all structures, systems and components (SSC) identified in the SMA success diagram were evaluated to have a HCLPF value >0.5g. The one exception is the non-safety related high pressure nitrogen bottle supply to the safety relief valve storage tanks. This nitrogen supply was assumed to be required to keep the safety relief valves open in the long term (>24 hours after the seismic event) when emergency depressurization is required to provide low pressure ECCS makeup to the reactor vessel.

   Regardless of the non-safety nitrogen supply exception, the IPEEE SMA-determined NMP2 plant HCLPF is 0.5g due to the three redundant options assessed for the RPV Inventory Control function of the NMP2 IPEEE SMA shutdown success paths: 1) HPCS; 2) RCIC or 3) ADS/LPI. The HPCS and RCIC options were both assessed to have a HCLPF greater than 0.5g. The other functions of the SMA shutdown success paths (Reactivity Control, RPV Pressure Control and Containment Pressure Control) were also assessed to each have a HCLPF value >0.5g.

   The 50th percentile HCLPF value of 0.42g (note: the SMA HCLPF of 0.5g is the 84th percentile value) determined in the NMP2 IPEEE. Section 3.2.4.3 of Reference 7 is selected as the representative plant HCLPF for this risk analysis as it is the value used in the SPRA portion of the NMP2 IPEEE. The uncertainty parameter for seismic capacity is represented by a composite beta factor (βc) of 0.46 with $\beta u = 0.44$ and $\beta R = 0.13$ (Reference 7).

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

4. Convolution to Determine SCDF: The estimation of SCDF in this calculation is performed by a mathematical convolution of the PGA-based seismic hazard curve and the NMP2 PGA-based plant HCLPF from the NMP2 IPEEE. This convolution estimation approach is a common analysis in approximating an SCDF for use in risk-informed decision making (e.g., it is commonly used in RICT seismic penalty calculations; the NRC used this approach in the GI-199 risk assessment (Reference 13) in absence of a current full-scope SPRA.

5. SLERF: The NMP2 SLERF for this risk evaluation is obtained by multiplying the calculated SCDF by an average seismic conditional large early release probability (SCLERP). The average SCLERP is estimated using information from both the NMP2 IPEEE SPRA and results from quantification of the NMP2 FPIE PRA model (Reference 14).

6. Consideration of S-LOOP: The analysis also assesses the incremental risk associated with seismic-induced LOOP that may occur from seismic events below the NMP2 seismic design basis. The analysis compares a convolution estimation of seismic-induced LOOP frequency with the random LOOP frequencies in the FPIE PRA. This analysis aspect and approach has been used in past RICT seismic penalty calculations.

Seismic Hazard and Intervals

The seismic hazard input developed from the NTTF 2.1 response letter and in the Seismic CDF-LERF Estimate for RICT is shown in Table E4-1. Several points have been interpolated to provide values at convenient seismic hazard points. A fitted curve was compared with the linear interpolation, it was observed that both methods bear the same total occurrence frequency, however the linear interpolation resulted in higher frequencies for the higher seismic magnitude, therefore the linear interpolation is selected to predict unknown points on the hazard curve. The mean fractile occurrence frequencies of Table E4-1 developed in Reference 12 are used in the calculations here; use of mean values is a typical and expected PRA practice. Table E4-2 shows the seismic interval magnitude range, representative g-level, and an acceptable common number of hazard intervals based on industry guidance provided in the EPRI SPRA Implementation Guide used in industry SPRAs (Reference 15).

The representative g-level for seismic intervals 1 through 7 is calculated as the square root of the product of the g-level values at the beginning and end of the range. For seismic interval 8, the representative g-level is estimated as 1.1 times the g-level at the beginning of the range since the interval has no upper limit. The interval frequency is calculated by subtracting the mean frequency associated with the g-interval (high) end point from the mean frequency associated with the g-interval beginning point. This is also common practice in industry SPRAs (Reference 15).

The first interval starts from 0.09g to capture the SSE (0.15g). The limiting HCLPF value for NMP2 (i.e., 0.42g) is sufficiently higher than the operating basis earthquake (OBE=0.075g). Given that the HCLPF represents the approximately 1%~5% likelihood of failure value, starting the first interval at 0.09g (i.e., slightly above) does not represent a significant impact for this risk evaluation.

Table E4-1: Seismic Hazard Data for Nine Mile Point 2[1]

AMPS(g)	MEAN	0.05	0.16	0.50	0.84	0.95
0.0005	5.59E-02	3.14E-02	4.37E-02	5.58E-02	6.93E-02	7.77E-02
0.001	4.18E-02	1.95E-02	3.05E-02	4.13E-02	5.42E-02	6.36E-02
0.005	1.16E-02	3.52E-03	6.36E-03	1.01E-02	1.62E-02	2.46E-02
0.01	4.91E-03	1.25E-03	2.13E-03	3.90E-03	6.83E-03	1.34E-02
0.015	2.68E-03	6.09E-04	9.93E-04	1.95E-03	3.73E-03	8.47E-03
0.03	8.04E-04	1.36E-04	2.13E-04	4.63E-04	1.08E-03	3.23E-03
0.05	3.00E-04	3.68E-05	6.00E-05	1.36E-04	3.95E-04	1.34E-03
0.075	1.32E-04	1.32E-05	2.29E-05	5.42E-05	1.72E-04	6.09E-04
0.1	7.31E-05	6.83E-06	1.21E-05	3.01E-05	9.37E-05	3.33E-04
0.15	3.11E-05	2.88E-06	5.50E-06	1.38E-05	4.07E-05	1.32E-04
0.3	6.92E-06	6.09E-07	1.36E-06	3.68E-06	9.65E-06	2.39E-05
0.5	2.16E-06	1.51E-07	3.95E-07	1.21E-06	3.33E-06	7.03E-06
0.75	8.08E-07	3.95E-08	1.21E-07	4.37E-07	1.32E-06	2.72E-06
1.	3.85E-07	1.27E-08	4.56E-08	1.95E-07	6.45E-07	1.36E-06
1.5	1.25E-07	2.04E-09	9.37E-09	5.35E-08	2.13E-07	4.70E-07
3.	1.36E-08	1.07E-10	3.84E-10	3.52E-09	2.13E-08	5.91E-08
5.	1.99E-09	4.56E-11	9.11E-11	3.42E-10	2.72E-09	9.24E-09
7.5	3.49E-10	3.79E-11	5.05E-11	9.51E-11	4.50E-10	1.69E-09
10.	9.02E-11	3.01E-11	4.01E-11	9.11E-11	1.42E-10	4.77E-10

The interpolated values in Table E4-1 (using linear straight line interpolation) were calculated performed for this analysis for use in calculation of the hazard interval frequencies for AMPS 0.09g, 0.175g, 0.2g, 0.25g, 0.35g, 0.4g, 0.6g, 0.7g, 0.8g, 0.9g, 1.1g, 1.2g, 1.3g, 1.4g. These specific PGA points are not listed in Reference 11. Interpolations performed only for PGA mean values for the base case evaluation and the sensitivity evaluation for 2.5 Hz.

Table E4-2: Seismic Hazard Intervals

ID	Seismic IE Interval Range (g, PGA)	Seismic IE Interval Representative Magnitude (g, PGA)	Interval Frequency

[1] Reproduced from Reference (Reference 11) Table A 1a. Mean and Fractile Seismic Hazard Curves for PGA at NMP2

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%G1	0.09 - 0.3	0.16	8.97E-05
%G2	0.3 - 0.5	0.39	4.76E-06
%G3	0.5 - 0.7	0.59	1.08E-06
%G4	0.7 - 0.9	0.79	5.24E-07
%G5	0.9 - 1.1	0.99	2.21E-07
%G6	1.1 - 1.3	1.20	1.04E-07
%G7	1.3 - 1.5	1.40	1.04E-07
%G8	>1.5	1.65	1.25E-07

Seismic Failure Probabilities

The seismic failure probability of the NMP2 limiting HCLPF for each hazard interval is calculated using the following fragility equations. These are the typical lognormal fragility equations used in most hazard PRAs (Reference 15).

Fragility (i.e., failure probability) = Φ (ln(A/Am)/ßc),
where
Φ is the standard lognormal distribution function
A is the g level in question,
Am is the median seismic capacity,
and the uncertainty parameters (betas) are related as follows:

ßc = (ßu^2 + ßr^2)^0.5

HCLPF and Am are related as follows:

Am = HCLPF / (exp -1.65(ßr + ßu))

The above fragility relationships are used in Reference 12 to determine the plant level seismic-induced failure probability as a function of seismic hazard interval. The following plant level HCLPF values from the NMP2 IPEEE are used: 0.42g PGA HCLPF (50^{th} percentile based HCLPF), ßr = 0.13 and ßu = 0.44 (which provide a composite ßc = 0.46). The median seismic capacity (Am) = 1.076.

With all parameters specified, the interval-specific failure probabilities are calculated as defined above. The interval-specific failure probabilities are shown in Table E4-3 for each interval. Note that in Table E4-3, the interval frequencies from Table E4-2 are repeated for convenience.

Table E4-3

Seismic-Induced Failure Probabilities for Each Hazard Interval for IPEEE

Seismic-Induced Plant Level Failure Probability as a Function of Seismic Hazard Interval							
Representative Seismic Magnitude per Interval (g, PGA)							
%G1	**%G2**	**%G3**	**%G4**	**%G5**	**%G6**	**%G7**	**%G8**
0.16	**0.39**	**0.59**	**0.79**	**0.99**	**1.20**	**1.40**	**1.65**
Seismic CCDP							
2.2E-5	1.3E-2	9.7E-2	2.5E-1	4.3E-1	5.9E-1	7.1E-1	8.2E-1
Seismic IE Frequency (/yr)							
8.97E-5	4.76E-6	1.08E-6	5.24E-7	2.21E-7	1.04E-7	1.04E-7	1.25E-7

Seismic Core Damage Frequency

The SCDF for each interval is the product of the seismic hazard interval frequency and the interval seismic failure probability. The total SCDF is the sum of all the hazard interval SCDFs. These results are shown in Table E4-4, which also shows the percentage of total SCDF for each interval. As shown in Table E4-4, the total estimated SCDF is 6.4E-7/yr.

In Table E4-3, the %G8 interval (i.e., seismic hazard internal >1.5g) is shown to have a CCDP of 0.82. This would indicate that the representative magnitude for the %G8 interval in not sufficiently high in order to reach a CCDP of closer to 1.0. A sensitivity study was performed to use a larger number of seismic hazard intervals to better represent the entire hazard curve (i.e., where the last seismic interval is calculated with a CCDP of 1.0). The sensitivity study supported that the total estimated SCDF would be less than 1% different than the SCDF value of 6.4E-7/yr shown in Table E4-4.

Table E4-4: Interval-Specific and Total SCDF

%G1	%G2	%G3	%G4	%G5	%G6	%G7	%G8	Total SCDF (/yr)
SDCF Contribution per Interval								
2.0E-9	6.3E-8	1.0E-7	1.3E-7	9.6E-8	6.1E-8	7.4E-8	1.0E-7	**6.4E-07**
SCDF % Contribution per Interval								
0.3%	9.8%	16.4%	20.9%	15.0%	9.6%	11.7%	16.2%	**100.0%**

Seismic Large Early Release Frequency (SLERF)

The NMP2 SLERF for this risk evaluation is obtained by multiplying the SCDF by the average seismic conditional large early release probability (SCLERP). An estimate of the average SCLERP is calculated using information from both the NMP2 IPEEE SPRA and from quantification results of the NMP2 FPIE PRA model. The NMP2 IPEEE used both the EPRI Seismic Margins Assessment (SMA) method and a seismic PRA for the seismic risk assessment. Although not required for the IPEEE, the NMP2 seismic PRA was performed to provide a quantitative perspective of the SMA. The NMP2 seismic PRA includes the following three fundamental technical areas:

- Seismic hazard analysis
- Response and fragility analysis
- SPRA systems and accident sequence analysis

Using information from the NMP2 IPEEE SPRA, the SLERF estimation approach used considers:

- The spectrum of Level 1 seismic-induced core damage accident sequence types and;
- Level 2 accident sequence progression and the influence of seismic-induced failures.

<u>Spectrum of Level 1 Seismic-Induced Core Damage Accident Sequence Types</u>

The estimation of an averaged SCLERP requires as an input the assessment of the contribution of different accident sequence types to seismic core damage frequency (SCDF). The contribution of various accident sequence types (or accident classes) to core damage frequency at a given plant is not necessarily the same between FPIE PRA and other hazard (e.g., seismic) PRAs. Although the NMP2 IPEEE SPRA was performed prior to the development of current SPRA methods and standards, that study does provide useful insights into seismic accident sequences. Therefore, the results from the NMP2 IPEEE SPRA are used here to define the spectrum of seismic-induced accident sequences types to SCDF.

Based on the information from the NMP2 IPEEE SPRA, the spectrum of Level 1 SCDF accident sequence types used in this analysis are defined as follows:

- <u>Seismic-LOOP with loss of containment cooling or other delayed loss of coolant injection</u>: These are seismic-induced loss of offsite power scenarios with RPV coolant makeup initially successful but containment cooling (e.g., RHR) is not successful (and thus coolant injection subsequently failed) or coolant injection fails late in the accident regardless of containment cooling status (e.g., failure of long term N2 supply fails RPV depressurization function). For the loss of containment cooling contributors to this category, adequate core cooling is subsequently failed (e.g., harsh environment in reactor building) due to primary containment overpressurization and failure.

- Seismic-LOOP with early loss of injection and **isolated** containment: These are seismic-induced loss of offsite power scenarios with RPV coolant injection failure at t=0, but successful containment isolation.

- Seismic-LOOP with early loss of injection and **unisolated** containment: These are seismic-induced loss of offsite power scenarios with RPV coolant injection failure at t=0 and failure of containment isolation. This includes early Station Blackout scenarios with failure to isolate containment or significant seismic failures (e.g., primary containment) that lead directly to core damage.

The NMP2 IPEEE states the following: *"Containment performance evaluations were included in the SMA studies which considered the primary containment structure, penetrations, piping and valves as well as the LOCAs outside containment. The HCLPF for these structures and components is judged to be much higher than the 0.5g plant HCLPF value…our judgement is that containment failure is dominated by station blackout scenarios with unisolated penetrations."*

At NMP2, the Drywell Floor Drain and Drywell Equipment Drain line penetrations have normally open MOVs as primary containment isolation valves (PCIVs) that fail as-is during a station blackout event. A proceduralized operator action is credited to manually, locally isolate the MOVs. Failure of the operator to isolate the MOVs during station blackout events is identified in the IPEEE as the dominant contributor to failure of containment isolation.

Based on the seismic results in the NMP2 IPEEE SPRA, the spectrum of NMP2 SCDF accident sequence types is summarized in Table E4-5, along with their percentage contributions to the IPEEE seismic CDF results.

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Table E4-5

Spectrum of SCDF Accident Sequences and Associated CLERP

SCDF (/yr)(1)	%SCDF	NMP2 IPEEE SPRA Level 1 Sequence Type	CLERP(2)	Comment
5.9E-8	46%	S-LOOP with late loss of containment cooling or injection	5.0E-2	The NMP2 Emergency Action Level procedure has directives (including Emergency Director discretion) that are modeled in the PRA with a high probability of issuance of a General Emergency well before primary containment failure and thus outside the "Early" timing criterion. However, the NMP2 FPIE PRA (Reference 14) includes a 5% probability that the General Emergency declaration is delayed and thus can result in an "Early" release. Use of 5E-2 for the CLERP is conservative in that this value does not account for primary containment failure location (i.e., if containment failure occurs in the wetwell airspace the release would be scrubbed and not a "High" magnitude release).
3.2E-8	25%	S-LOOP with early loss of injection and isolated containment	1.2E-2	Based on NMP2 FPIE PRA (i.e., LOOP with no injection at t=0) accidents with no AC recovery and no coolant injection recovery in Level 2 PRA. NMP2 Mark II primary containment does not have steel shell liner with air gap (such as in Mark I containments) and thus likelihood of a "High" magnitude release for an unmitigated core damage accident is lower in comparison to a Mark I containment design.

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SCDF (/yr)(1)	%SCDF	NMP2 IPEEE SPRA Level 1 Sequence Type	CLERP(2)	Comment
3.7E-8	29%	S-LOOP with early loss of injection and unisolated containment. Scenarios modeled as leading to core damage (e.g., Station Blackout with no credit for offsite AC power recovery) and failure to isolate containment.	1	Section 3.2 of the NMP2 IPEEE identifies that this category (i.e., Early unisolated containment) has a CDF of 1.6E-7/yr based on assuming the SMA HCLPF of 0.5g for failure of all safety related equipment in the SMA success path and leading directly to core damage. However, Section 3.2 further states that a best estimate is 3.7E-8/yr for seismic induced Station Blackout events leading to the early unisolated containment end state. Containment performance evaluations were included in the NMP2 IPEEE SMA studies, which considered the primary containment structure, penetrations, piping and valves, as well as LOCAs outside containment. The HCLPF for these structures and components is judged to be much higher than 0.5g, such that the contribution to the unisolated containment end state is not significant. Therefore, the best estimate of 3.7E-8/yr is used for the seismic CLERP evaluation. In addition, the CLERP is likely conservative. The operator action (i.e., ZQ-HEP-IND-ZIS03) to manually, locally isolate normally open containment isolation MOVs (e.g., drywell equipment and floor drain MOVs), that fail as-is during station blackout scenarios, is modeled in the FPIE with a Human Error Probability (HEP) of 0.3 (Reference 16). Given the potential adverse seismic impacts on Performance Shaping Factors (PSFs) (e.g., cue, travel time, stress level), the HEP for failure to manually, locally isolate the MOVs during seismic station blackout scenarios is conservatively assumed to be 1.0 for the entire range of seismic hazard magnitudes.
Sequence-Weighted Averaged CLERP:			**0.32**	**Sum of (%SCDF x CLERP) over all sequences**

Notes to Table E4-5:

(1) Mean annual Seismic CDF using EPRI hazard curve and information from Section 3.2 of NMP2 IPEEE (Reference 7).

(2) Seismic conditional large early release probability values are based on insights from similar sequences as modeled in the NMP2 FPIE PRA (Reference 14), per the discussion in the Comment column.

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The next step in the estimation of an averaged seismic CLERP is to estimate the CLERP for each SCDF accident sequence type. A given accident sequence type may not result in a core damage event until well after the PRA "Early" release time frame (defined in the NMP2 FPIE PRA as ≤ 7 hours from the time of the cue for a General Emergency declaration). Conversely, some accident sequence types would, by PRA convention, be modeled directly as a LERF, such as a station blackout scenario with failure to manually isolate containment isolation valves.

The sequence-weighted average SCLERP over the SCDF accident sequence contributions and assigned SCLERPs is estimated as 0.32. As discussed above, this SCLERP estimate contains conservatisms.

This average CLERP estimation includes consideration of seismic-induced failure of the primary containment structural boundary. As discussed earlier, the NMP2 IPEEE states the following: *"Containment performance evaluations were included in the SMA studies which considered the primary containment structure, penetrations, piping and valves as well as the LOCAs outside containment. The HCLPF for these structures and components is judged to be much higher than the 0.5g plant HCLPF value…our judgement is that containment failure is dominated by station blackout scenarios with unisolated penetrations."* Therefore, failure of the primary containment or the containment boundary (e.g., penetrations and piping) are judged to have a non-significant contribution to the seismic CLERP estimate.

Based on the information in Table E4-5, an estimate of CLERP based on the NMP2 IPEEE SPRA results is 0.32, i.e., seismic LERF is equal to 32% of the seismic CDF estimate). To account for potential uncertainties in the CLERP calculation for use in RICT calculations for NMP2, a CLERP of 0.5 will be used as a conservative value to provide additional safety margin for use in the seismic LERF "penalties" for the RICT calculations.

Therefore, a conservative estimate of SLERF is:

SLERF = 6.4E-7/yr (SCDF) x 0.5 (CLERP) = 3.2E-7/yr.

The estimated SLERF will be used for the base case SLERF value for RICT calculations that applies when the primary containment is inerted. If a RICT is being entered during a period when the primary containment is de-inerted, a different SLERF penalty of 6.4E-07/yr (SCDF = SLERF, CLERP = 1.0) will be applied to address the increased potential for hydrogen deflagration events in the primary containment. This is deemed conservative since the NMP2 Level 2 FPIE PRA (Reference 17) credits steam inerting in the primary containment with an estimated 0.5 failure probability that would mitigate the hydrogen deflagration event. Given the uncertainty in the steam inerting value of 0.5 and the small time frame for potential de-inerted conditions, a conservative assumption for CLERP of 1.0 will apply when the containment is de-inerted.

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Application of SLERF in RICT Calculations

Conservatism in the RICT process derives from the proposed approach to apply the total estimated annual seismic LERF as a delta SLERF in each RICT calculation, regardless of the duration of the completion time. The total estimated annual seismic CDF and LERF will be applied starting at time zero for each RICT calculation.

3.1.1 RICT Conclusion

Estimates of SCDF and SLERF have been derived for use in the NMP2 TSTF-505 program. The estimates are intended to be treated as conservative values in the RICT calculations for that program.

Containment inerted:

For this case a SCDF of 6.4E-7 and a SLERF penalty of 3.2E-7/yr will be used.

> Plant level HCLPF = 0.42g PGA with ßc = 0.46 will be used for the calculation of SCDF with an average SCLERP of 0.5 (i.e., SCDF = 6.4E-7/yr, SLERF = 3.2E-7/yr).

Containment de-inerted:

For this case a SCDF of 6.4E-7 and a SLERF penalty of 6.4E-7/yr will be used.

> SCLERP = 1.0 (SCDF = SLERF = 6.4E-7/yr)

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4 Extreme Winds Analysis

Section 3.3.1 of the USAR (Reference 18) documents that the NMP2 Category I buildings are designed to withstand a fastest mile wind velocity of 90 miles per hour. USAR Section 3.3.2 and associated Table 3.2-1 document that key equipment and structures are designed to withstand tornados with a maximum rotational velocity of 290 mph, a maximum translational velocity of 70 mph, a maximum external pressure drop of 3 psi, and a maximum rate of pressure drop of 2 psi/sec. The maximum resultant wind velocity is 360 mph. USAR Table 3.5-21 lists the postulated missiles used in the NMP2 design basis analysis. Hurricane winds are not a concern at NMP2, due to its inland location. Therefore, the NMP2 tornado design is bounding for high winds.

Section 5.1.4 of the IPEEE evaluates the risk associated with tornado missiles. The result of the IPEEE analyses is a tornado missile CDF less than 1.0E-07 per year. Tornado hazard frequencies used in evaluating and dispositioning potential tornado vulnerabilities in the IPEEE bound more current tornado hazard frequencies (i.e., from NUREG/CR-4461, Revision. 2 (Reference 19)). Based on the data in NUREG/CR-4461, Revision 2, design basis tornado wind speeds have a frequency less than 1E-7/yr at the NMP2 site.

Subsequent to the IPEEE, NMP2 performed tornado missile protection (TMP) evaluations between 2015 and 2017. As part of the NRC issued Regulatory Issue Summary (RIS) 2015-06 (Reference 21) and Enforcement Guidance Memorandum (EGM) 15-002 (Reference 22), all licensees of operating plants were reminded of their obligations to comply with their licensing basis and regulatory commitments to protect safe shutdown equipment from tornado generated missiles. As a result of the TMP project, a list of components was developed that may be impacted by tornado missiles, either a direct missile path or an indirect path due to cable routing (Reference 23).

Although several SSCs were identified as potentially vulnerable to tornado missiles, they were determined to meet the NMP2 design and licensing bases (Reference 18). However, these potentially vulnerable SSCs could contribute to tornado missile risk. Therefore, the risk associated with the identified SSCs remaining unprotected from tornado missiles was evaluated (Reference 20). Only one of the unprotected SSCs is included in the NMP2 internal events PRA; it was conservatively estimated that the likelihood of a tornado missile strike on that SSC was much less than 1E-6/yr, which results in an insignificant contribution to CDF. The low risk from tornado missiles based on current TMP evaluations is consistent with the IPEEE results.

Extreme Winds Analysis Summary

The NMP2 plant design basis meets the requirements of the NRC Standard Review Plan (SRP) with regards to tornado wind and missile hazards. Design basis tornado wind speeds have a frequency less than 1E-7/yr at the NMP2 site, based on data from NUREG/CR-4461, Revision 2. Tornado missile CDF documented in the IPEEE is less than 1E-7/yr; a review of more recent TMP

evaluations concludes that CDF is much less than 1E-6/yr. Therefore, high winds and tornado missiles can be screened from consideration in risk informed applications.

Configuration Specific Considerations

As noted in the site-specific evaluation presented above, there are no configuration specific considerations related to the screening assessment for high winds / tornados provided above for NMP2.

5 External Flooding Analysis

The evaluation of the impact of the external flooding hazard at the site was updated as a result of the NRC's post-Fukushima 50.54(f) Request for Information. The station’s flood hazard reevaluation report (FHRR) was submitted to NRC for review on March 12, 2013 (Reference 24). The results indicate that all flood causing mechanisms, except Local Intense Precipitation (LIP), are bounded by the current licensing basis (CLB) and do not pose a challenge to the plant.

LIP was reevaluated and found to produce water surface elevations (WSEs) less than that of the CLB elevation of 262.5 ft at NMP Unit 2. However, the duration for which the flood is estimated to have an impact on the site was estimated to be 52.5 hours in the FHRR and the CLB estimated only 20 hours. Therefore, the impact considering the increased duration of the flooding event was evaluated in WH-C-002, Rev. 0, "Impact of NMP2 Areas Housing Safe Shutdown Equipment due to water ingress from Probable Maximum Flood" (Reference 25). Flooding volumes were calculated based on the revised flood inundation time, in-leakage, and building drainage features in Calculation WH-C-002. This analysis did not assume any temporary barriers were installed prior to the event starting and all doors were assumed in their normal positions. The document concludes that a LIP event will not cause enough water to enter buildings with safety-related (SR) SSCs or accumulate to a depth that affects any of the SR SSCs.

As a measure of defense-in-depth, the mitigating strategies assessment (MSA) for flooding confirms that FLEX will remain deployable during a LIP event with the installation of temporary flood barriers prior to the start of rainfall. Once NMP2 receives a warning of a rainfall of greater than 6 inches in the next 24 hours or an intensity of more than 2 in/hr, station procedures direct the installation of temporary flood protection measures to allow the FLEX support guidelines to be implemented. FLEX is capable of providing alternate means of maintaining key safety functions for the duration of the flood event and further reducing the overall risk posed by external flooding to NMP2.

Disposition for RICT Program

All non-LIP flooding mechanisms were considered bounded by the plants CLB and design basis. These mechanisms will not produce a flood with a WSE high enough to impact any SSCs at NMP Unit 2.

Flooding from LIP will similarly not challenge any safety functions at NMP Unit 2. The maximum WSE is calculated to be 1.4 ft above the building finished floor elevation for a period of 52.5 hours; however, WH-C-002, Rev. 0, demonstrated that water in-leakage will not accumulate enough in any buildings to impact any SR SSCs. Therefore, LIP and all other flooding mechanisms are screened from inclusion in the RICT calculations due to the limited impact presented from external flooding.

As a measure of defense-in-depth, the mitigating strategies assessment (MSA) for flooding confirms that FLEX will remain deployable during a LIP event with the installation of temporary flood barriers prior to the start of rainfall.

Configuration Specific Considerations

There are no configuration specific considerations related to the screening assessment provided above for NMP2.

External Flooding Assessment Results Summary

Based on this analysis of the external flooding hazard for NMP2, the hazard has negligible impact on the RICT program.

6 Evaluation of External Event Challenges and IPEEE Update Results

This section provides an evaluation of other external hazards. The results of the assessment of these hazards is provided in Table E4-8. Table E4-9 provides the summary criteria for screening of the hazards listed in Table E4-8.

Hazard Screening

The IPEEE for NMP2 Units 1 and 2 provides an assessment of the risk to NMP2 associated with these other external hazards. Additional analyses have been performed since the IPEEE to provide updated risk assessments of various hazards, such as aircraft impacts, industrial facilities and pipelines, and external flooding. These analyses are documented in the USAR. Table E4-8 reviews and provides the bases for the screening of external hazards, identifies any challenges posed, and identifies any additional treatment of these challenges, if required. The conclusions of the assessment, as documented in Table E4-8, assure that the hazard either does not present a design-basis challenge to NMP2, or is adequately addressed in the PRA.

In the application of Risk-Informed Completion Times, 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 evaluation for NMP2 has been performed accounting for such configuration-specific impacts. The process involves several steps.

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As a first step in this screening process, hazards that screen for one or more of the following criteria (as defined in Table E4-9) 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 is 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.

Seismic-Induced Loss of Offsite Power Challenges

As part of the evaluation of seismic risk for the TSTF-505 application, it is also necessary to evaluate any incremental risk associated with challenges to the facility that do not exceed the design capacity (because the preceding evaluation examines the seismic risk from the hazard above the SSE of 0.15g). This incremental risk is associated with loss of offsite power (LOOP). The methodology for computing the seismically-induced LOOP frequency is to convolve the NMP2 mean seismic hazard curve with the offsite power fragility. The NMP2 seismic hazard curve is as described in Table E4-1.

Table E4-6 provides the mean seismic hazard data and the LOOP failure probability for each seismic interval based on the fragility of offsite power. Note that in this table the lower bound g-level is selected as 0.03g. This value is less than to the operating basis earthquake (OBE=0.075g) for NMP2. OBE is equal to the acceleration at which the plant is designed to continue to operate, as and is a convenient point for this evaluation because it is a data point on the tabulated hazard curve data. Inclusion of g-levels lower than this, although higher in frequency, would not significantly affect the conclusions of this evaluation because the seismic induced loss of offsite power frequency is much lower than the unrecovered non-seismic LOOP frequency from the FPIE PRA model.

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The failure probabilities for LOOP are represented by failure of ceramic insulators in the power distribution system, based on the following fragility data from the NMP2 IPEEE:

$$Am = 0.3g; \beta R = 0.13, \beta U = 0.44$$

Given the mean frequency and failure probability for each seismic interval, it is straightforward to compute the estimated frequency of seismically induced loss of offsite power for the NMP2 site by taking the product of the interval frequency and the offsite power failure probability (which is calculated as explained in Section 3.2). As shown in Table E4-6, the total seismic LOOP frequency is the sum of the individual interval frequencies, or approximately 1.4E-5/yr. Note that this overstates the beyond-design challenge frequency but is conservative for this purpose. In Table E4-6, the %G1 interval (i.e., seismic interval from 0.09g to 0.3g) represents approximately 61% of the total seismic induced LOOP frequency. The %G2 interval (0.3g to 0.5g) represents approximately 24% of the total seismic induced LOOP frequency. The frequency contribution for %G1 interval could likely be reduced if it was divided into 2 or more intervals. However, the splitting of interval %G1 is not necessary for this evaluation since the total seismic LOOP frequency derived in Table E4-6 is sufficiently low compared to the random (i.e., non-seismic) LOOP frequency.

The FPIE PRA models LOOP from plant-centered, switchyard-centered, grid-related, and weather-related events. Based on the NMP2 FPIE PRA, the total frequency of unrecovered loss of offsite power (i.e., the sum of the frequency multiplied by the non-recovery probability at 24 hours over these LOOP events), is 6.8E-4/yr, as shown in Table E4-7.

The seismically-induced (unrecoverable) LOOP frequency is therefore approximately 1.8% of the total unrecovered LOOP frequency already addressed in the FPIE PRA. This evaluation is conservative because the purpose of the comparison is to support that the seismic induced LOOP frequency below the design basis (i.e., SSE of 0.15g) is less than the random LOOP frequency. If the evaluation was limited to the seismic hazard curve below the design basis, the calculated seismic induced LOOP frequency would be much smaller than when evaluating the seismic induced LOOP frequency for the entire seismic hazard curve. Despite this conservatism, the calculated seismic induced LOOP frequency is judged to be a reasonably small fraction such that it will not significantly impact the RICT Program calculations and it can be omitted. It was observed that the use of the 8 hazard intervals approach is conservative for this S-LOOP estimate by about 10-15% compared to if many more intervals of thinner width were used.

Table E4-6: NMP2 Seismic LOOP Frequency Estimate

Seismic Interval (g)	Interval Representative g Level	Interval Frequency (/yr)	Offsite Power Failure Prob.	Seismic Interval LOOP Frequency
0.03 - 0.09	0.16	8.97E-05	9.47E-02	8.50E-06
0.09 - 0.3	0.39	4.76E-06	7.11E-01	3.38E-06
0.3 - 0.5	0.59	1.08E-06	9.31E-01	1.01E-06
0.5 - 0.7	0.79	5.24E-07	9.83E-01	5.15E-07
0.7 - 0.9	0.99	2.21E-07	9.96E-01	2.20E-07
0.9 - 1.1	1.20	1.04E-07	9.99E-01	1.04E-07
1.1 - 1.3	1.40	1.04E-07	1.00E+00	1.04E-07
>1.5	1.65	1.25E-07	1.00E+00	1.25E-07
Total Seismic LOOP Frequency =				**1.4E-05**

Table E4-7: NMP2 FPIE PRA Loss of Offsite Power (LOOP) Non-Recovery Frequency

LOOP Contributor	LOOP Contributor Frequency Prior Mean (/Crit Yr) (1),(2)	Probability of Non-Recovery by 24 hours (3),(4)	24-hr Non-Recovered LOOP Frequency (Interpolated)
Plant-Centered	1.71E-03	1.11E-03	1.8E-06
Switchyard-Centered	1.20E-02	2.25E-03	2.5E-05
Grid-Related	2.51E-02	3.42E-03	8.1E-05
Weather-Related	5.35E-03	1.14E-01	5.7E-04
Total			**6.8E-04**

(1) Section 3.4.2 of N2-PRA-001 - NMP2 Initiating Events NB (Reference 26)

(2) Values are per N2-PRA-001, Table M-5

(3) Use non-recovery factors at 24 hours Table 3-6 in N2-PRA-001.

(4) The posterior mean values could be from "events per critical year" to "events per year" by multiplying the values by an average criticality factor for NMP2. The average criticality factors for NMP2 is 0.94. "Events per critical year" is used conservatively.

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Table E4-8 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**NINE MILE POINT 2 Response**
Aircraft impacts	A direct or indirect (i.e. skidding impact) collision of a portion of or an entire aircraft with one or more structures at or in the area surrounding the plant site.	Y	PS2 PS4	Per USAR Section 2.2.3.1.7, the nearest air corridor is approximately 22.5 km (14 mi) east of the site (Section 3.5.1.6). There are only two airfields between the 8-km (5-mi) and 24-km (15-mi) radii of the site; the Lakeside Airpark and Oswego County Airport are about 12 km (7.5 mi) and 19 km (12 mi) south of the site, respectively. The aircraft approaches to these airports are not near the plant site. The general aviation movements at these airports total approximately 1,460/yr and 19,900/yr, respectively. The annual movements are below the critical number at which a probability analysis for aircraft accidents would be required according to RG 1.70 (Reference 30). Therefore, the probability of aircraft crashing into the site is considered to be remote, and airplane crashes need not be considered design basis events. Similarly, for helicopter operations to and from the site, the probability of a helicopter crash resulting in radiological releases in excess of 10CFR100 guidelines has been conservatively estimated to be approximately 1 x 10-6, using the methodology of NRC SRP 3.5.1.6 (Reference 31). In accordance with SRP 2.2.3 (Reference 32), additional qualitative arguments could be made which would substantially lower this probability to less than about 10-7 per year. This satisfies the requirements of RG 1.70 such that helicopter crashes need not be considered as design basis events.

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Table E4-8
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	NINE MILE POINT 2 Response
				Based on this review, the Aircraft Impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Avalanche	A rapid flow of a large mass of accumulated frozen precipitation down a sloped surface.	Y	C3	The New England location of NMP2 station along Lake Ontario precludes the possibility of an avalanche. Based on this review, the Avalanche impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Biological events	The accumulation or deposition of vegetation or organisms (e.g., zebra mussels, clams, fish) on an intake structure or internal to a system that uses an intake structure.	Y	C3 C5	Hazard is slow to develop and can be identified via monitoring and managed via standard maintenance process. Actions committed to and completed by NMP2 in response to Generic Letter 89-13 provide on-going control of biological hazards. These controls are described in Exelon procedure ER-AA-340, "GL 89-13 Program Implementing Procedure (Reference 33). Based on this review, the Biological Event impact hazard can be considered to be negligible.

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Table E4-8
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	NINE MILE POINT 2 Response
				There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Coastal erosion	The wearing away of a shoreline due to wave action, tidal currents, wave currents, drainage, or winds.	Y	C3	Per USAR Section 2.5.1.1.5, a dike was built which prevents waves from reaching unit structures thereby eliminating the hazard of shorline erosion at the site. Based on this review, the Coastal Erosion impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Drought	An extended period of months or years when a region experiences a deficiency in its surface or underground water supply	Y	C5	Drought is a slowly developing hazard allowing time for orderly plant reductions, including shutdowns. Based on this review, the Drought impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

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Table E4-8 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**NINE MILE POINT 2 Response**
External Flooding	Accumulation of excessive water on the station grounds from various sources including Local Intense Precipitation and Snow Accumulation	Y	C1 PS2	See information in Section 5 of this Enclosure.
Extreme Wind or Tornado	Excessive winds, straight-line or tornadic	Y	C1 PS2 PS4	See information in Section 4 of this Enclosure.
Fog	Water droplets suspended in the atmosphere at or near the Earth's surface that limit visibility.	Y	C1	The principal effects of such events (such as freezing fog) would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for NMP2. Based on this review, the Fog impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation
Forest or Range Fire	Fires originating from outside the plant site boundary that are caused by the uncontrolled combustion of vegetation (e. g. , trees, grasses, brush, etc.)	Y	C3	Forest fires were screened in the IPEEE. The site landscaping and lack of forestation prevent such fires from posing a threat to NMP2 station. Per USAR 2.2.3.1.4, the site is sufficiently cleared in areas adjacent to the plant that forest or brush fires pose no safety hazards.

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Table E4-8
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	NINE MILE POINT 2 Response
				Based on this review, the Forest or Range Fire impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Frost	A thin layer of ice crystals that form on the ground or the surface of an earthbound object when the temperature of the ground or surface of the object falls below freezing.	Y	C1	The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for NMP2. Based on this review, the Frost impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Hail	Showery precipitation in the form of irregular pellets or balls of ice.	Y	C1	The principal effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for NMP2. Based on this review, the Hail impact hazard can be considered to be negligible.

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Table E4-8 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**NINE MILE POINT 2 Response**
				There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
High summer temperature	High abnormal ambient temperatures.	Y	C1 C5	The principal effects of such events would result in elevated lake temperatures which are monitored by station personel. Should the ultimate heat sink temperature exceed the NMP2 Technical Specification 3.7.1 temperature limit (Reference 34), an orderly shutdown would be initiated. Plant is designed for this hazard. Associated plant trips are rare and are covered in the definition of another event in the PRA model (e.g., transients, loss of condenser). Based on this review, the High Summer Temperature impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
High tide, Lake Level, or River Stage	The periodic maximum rise of sea level resulting from the combined effects of the tidal gravitational forces exerted by the Moon and Sun and the rotation of the Earth.	Y	C3 C4 C5	Per USAR 2.4.1, dams on the St. Lawrence River, the outlet to Lake Ontario, under the authority of the International St. Lawrence River Board of Control, are used to regulate the lake level. The low limit is set for el 74.37 m (244 ft) on April 1 and is maintained at or above that elevation during the entire navigation season (April 1 to November 30). The upper limit of the lake level is el 75.59 m (248 ft). Per USAR

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Table E4-8
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	NINE MILE POINT 2 Response
				2.4.4, potential dam failures have been considered in plant design. Tide magnitudes amount to less than 2.5 cm (1 inch). Based on this review, the High Tide, Lake Level, or River Stage impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Hurricane	An extremely large, powerful, and destructive storm resulting in strong winds, excessive rainfall, high waves, storm surge, and tornados.	Y	C3	The location of NMP2 along Lake Ontario precludes the possibility of a hurricane. Based on this review, the Hurricane impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Ice cover	The accumulation of frozen water on bodies of water (e.g., lakes, rivers, etc.) or on structures, systems, and components.	Y	C1 C4	The effects of such events would be to cause a loss of off-site power and are addressed in the weather-related Loss of Offsite Power initiating events in the internal events PRA model for NMP2. In addition, per USAR 2.4.7, the Unit 2 design incorporates features to minimize the potential for cooling water blockage by ice.

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Table E4-8
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	NINE MILE POINT 2 Response
				Based on this review, the Ice Cover impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Industrial or military facility accident	An accident at an offsite industrial or military facility such as a release of toxic gases, a release of combustion products, a release of radioactivity, an explosion, or the generation of missiles.	Y	C1 C3 PS2	There are no chemical plants, refineries, military bases, or underground gas storage facilities within 8 km (5 mi) of the plant per USAR Section 2.2.1. There are several hazardous products or materials regularly manufactured, stored, used or transported within 5 miles of the site. There are five industrial facilities that are within 5 miles of the station per Table 2.2-3 of the USAR. Hazardous chemicals used and/or stored by manufacturers within five miles of the plant were evaluated and determined to either screen from further evaluation or were determined to meet the acceptance criteria associated with Control Room operator protection as discussed in NMP2 USAR, Section 2.2. Based on this review, the Industrial or Military Facility Accident impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

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Table E4-8 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**NINE MILE POINT 2 Response**
Internal Flooding	Excessive water accumulation internal to the station buildings	N/A	None	The NMP2 Internal Events PRA includes evaluation of risk from internal flooding events.
Internal Fire	Fire events that are internal to the station buildings	N/A	None	The NMP2 Internal Fire PRA includes evaluation of risk from internal fire events.
Landslide	A rapid flow of a large mass of earth, rock, or material other than accumulated frozen precipitation down a sloped surface.	Y	C3	Plant site is located on level terrain and is not subject to landslides. Based on this review, the Landslide impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Lightning	An electrical discharge from a cloud to the ground or Earth-bound object.	Y	C1 C4	Lightning strikes are not uncommon in nuclear plant experience. They can result in losses of off-site power or surges in instrumentation output if grounding is not fully effective. The latter events often lead to reactor trips. Both events are incorporated into the NMP2 internal events model through the incorporation of generic and plant specific data. Based on this review, the Lightning impact hazard can be considered to be negligible.

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Table E4-8 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**NINE MILE POINT 2 Response**
				There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Low Lake Level or River Stage	A decrease in the water level of the lake or river used for power generation.	Y	C5	Per USAR 2.4.1, dams on the St. Lawrence River, the outlet to Lake Ontario, under the authority of the International St. Lawrence River Board of Control, are used to regulate the lake level. The low limit is set for el 74.37 m (244 ft) on April 1 and is maintained at or above that elevation during the entire navigation season (April 1 to November 30). The upper limit of the lake level is el 75.59 m (248 ft). Per USAR 2.4.4, potential dam failures have been considered in plant design. Tide magnitudes amount to less than 2.5 cm (1 inch). Based on this review, the Low Lake Level or River Stage impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Low Winter Temperature	Low abnormal ambient temperatures.	Y	C1 C5	The principal effects of such events would be to cause a loss of off-site power. These effects would take place slowly allowing time for orderly plant reductions, including shutdowns. At worst, the loss of off-site power events would be subsumed into the base PRA model results. Based on this review, the Low Winter Temperature impact hazard can be considered to be negligible.

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Table E4-8 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**NINE MILE POINT 2 Response**
				There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Meteorite or Satellite Impact	A meteoroid or artificial satellite that releases energy due to its disintegration in the atmosphere above the Earth's surface, direct impact with the Earth's surface, or a combination of these effects.	Y	PS4	The frequency of a meteor or satellite strike is judged to be so low as make the risk impact from such events insignificant. This hazard also was reviewed as part of the IPEEE submittal and screened based on low frequency of occurrence. Based on this review, the Meteorite or Satellite Impact impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Pipeline accident	An accident involving the rupture of a pipeline carrying hazardous materials or toxic gases.	Y	C1	Per USAR Section 2.2.1, there are two natural gas pipelines within 5 miles of the site. The nearest gas pipeline is over 3.2 km (2 mi) from NMP2. There is not significant hazard from explosions involving these pipelines that could interact with the plant. Based on this review, the Pipeline Accident impact hazard can be considered to be negligible.

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Table E4-8 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**NINE MILE POINT 2 Response**
				There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Release of Chemicals in Onsite Storage	An onsite accident involving the storage or handling of hazardous materials such as a release of toxic gases, a release of combustion products, a release of radioactivity, an explosion, or the generation of missiles. In this context, an onsite release of radioactivity is assumed to be associated with low-level radioactive waste.	Y	C4 PS1	The impact of releases of hazardous materials stored on-site was evaluated in the IPEEE submittal and updated in NMP2's USAR, Section 2.2.3.1.3 Additionally, a Control Room Envelope (CRE) Habitability Program is required by plant Technical Specification 5.5.13; thus these are periodically reevaluated in accordance with the specification. Other spills were determined to have no adverse effects on operation of plant equipment. Per the CRE Habitability Program hazardous chemical evaluations have been performed for all of the chemicals stored onsite. The impact of releases of chemicals in onsite storage do not pose a risk to the site. Chemicals at the NMP2 site were analyzed in accordance with the guidance provided in RG 1.78. Most of the chemical screened due to small quantities in small containers whereas the rest of the chemicals were analyzed assuming the maximum control room intake (1300 cfm in two train pressurization mode) is unfiltered. The results showed that all the chemicals were well below the toxicity limits. Therefore, none of the chemicals post a threat to control room operators at NMP2.

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Table E4-8
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	NINE MILE POINT 2 Response
				See also "Transportation Accidents." Based on this review, the Release of Chemicals in Onsite Storage impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
River diversion	The redirection of all or a portion of river flow by natural causes (e.g. a riverine embankment landslide) or intentionally (e.g. power production, irrigation, etc.).	Y	C3	The location of NMP2 along Lake Ontario precludes the possibility of a river diversion. Based on this review, the River Diversion impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Sand or Dust Storm	A strong wind storm with airborne particles of sand and dust.	Y	C1 C5	The plant is designed for such events. More common wind-borne dirt can occur but poses no significant risk to NMP2 given the robust structures and protective features of the plant. Based on this review, the Sand or Dust Storm impact hazard can be considered to be negligible.

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Table E4-8 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**NINE MILE POINT 2 Response**
				There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Seiche	An oscillation of the surface of a landlocked body of water, such as a lake, that can vary in period from minutes to several hours.	Y	C1 C3	Per the IPEEE Table 5.4-1, the principal effects of severe weather storms (including seiche flooding) would be to cause a loss of off-site power. In addition, seiche flooding was evaluated in the USAR Section 2.4.5. Water surface setup and seiche are produced by winds and atmospheric pressure gradients. These short-term lake fluctuations are generally less than 0.6m (2 ft) in amplitude. Based on this review, the Seiche impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Seismic activity	A sudden release of energy from the Earth's crust resulting in strong ground motion.	N	None	See information in Section 3 of this Enclosure.
Snow	The accumulation of snow on structures, systems, and components	Y	C1 C4 C5	This hazard is slow to develop and can be identified via monitoring and managed via normal plant processes. Potential flooding impacts covered under external flooding.

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Table E4-8 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**NINE MILE POINT 2 Response**
				Based on this review, the Snow impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Soil shrink-swell	The relative change in volume of the soil as a result of the type of soil and the amount of moisture.	Y	C1 C5	The potential for this hazard is low at the site, the plant design considers this hazard and the hazard is slow to develop and can be mitigated. Based on this review, the Soil Shrink-Swell Consolidation impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Storm surge	An abnormal rise in sea level accompanying a hurricane or other intense storm, whose height is the difference between the observed level of the sea surface and the level that would have occurred in the absence of the intense storm.	Y	C3 C4	The location of NMP2 along Lake Ontario precludes the possibility of a sea level driven storm surge. Potential flooding impacts by water levels of Lake Ontario are covered under external flooding. Based on this review, the Storm Surge impact hazard can be considered to be negligible.

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Table E4-8 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**NINE MILE POINT 2 Response**
				There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Toxic Gas	An onsite accident involving the storage or handling of hazardous materials such as a release of toxic gases, a release of combustion products, a release of radioactivity, an explosion, or the generation of missiles. In this context, an onsite release of radioactivity is assumed to be associated with low-level radioactive waste.	Y	C4	USAR Section 2.2.3.1.3 discusses toxic gas. There is no onsite storage of chlorine; sodium hypochlorite/sodium bromide biocide system is used, thus eliminating an onsite chlorine hazard. In addition, there is no possibility of an accident that could lead to the formation of flammable clouds in the vicinity of NMP2 because (1) there is no chemical plant in the vicinity; (2) no gas pipeline passes the station that presents an explosion hazard; and (3) no liquefied gases are transported in the vicinity. Per the IPEEE, the bounding analysis showed that these accidents do not significantly contribute to the plant risk. See also Transportation Accidents. Based on this review, the Toxic Gas impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.

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Table E4-8
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	NINE MILE POINT 2 Response
Transportation accidents	An accident involving damage to a land-based or marine vehicle transporting hazardous materials that may result in a release of toxic gases, a release of combustion products, or an explosion.	Y	C3 C4 PS2 PS4	The impact of transportation accidents was evaluated in the IPEEE and screened out utilizing NMP2 compliance with the SRPs. Per the USAR: Major transportation facilities are shown on USAR Figure 2.2-1. USAR Section 2.2.3 includes an evaluation of potential accidents, and their effects on the plant or plant operation. Types of accidents considered include explosions, flammable vapor clouds, toxic chemicals, fires, collisions with intake structures, and liquid spills. Based on a comprehensive survey of industries within a 10-km (6.2-mi) radius of Unit 2, the nearest highway on which explosive materials can be transported is Route 104, which is about 6.2 km (3.9 mi) from safety-related structures. This separation distance far exceeds the safe distance for truck traffic (approximately 548.6 m, 1,800 ft) given on Figure 1 of RG 1.91. In discussions with Conrail, it was determined that no explosive or flammable materials are transported to the Oswego terminal of the rail line between Oswego and Mexico, NY. In any event, the distance from this rail line to Unit 2 is much greater than the safe distance for rail traffic given in RG 1.91. Since the nearest commercial shipping lanes on Lake Ontario are more than 10 km (6.2 mi) from Unit 2 (according to the U.S. Coast Guard), potential explosions on a ship or barge is

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Table E4-8
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	NINE MILE POINT 2 Response
				well beyond the radius of the peak incident pressure of 1 psi as given in RG 1.91. For the Nine Mile Point site, sources of potential toxic chemical hazards include four stationary and two transportation sources within 8 km of the site. USAR Table 2.2-7 lists the chemicals associated with each source along with their quantities and distances from the Unit 2 control room air intake. The effect of an accidental release of each of the chemicals described in the previous section on control room habitability was evaluated by calculating vapor concentrations inside the control room as a function of time following the accident. This calculation is performed using the conservative methodology outlined in NUREG-0570 and utilizing the assumptions described in RG 1.78. The results of the analysis are summarized in Table 2.2-8, which indicates that none of the toxic chemicals evaluated have the potential to incapacitate the Control Room Operators. Based on this review, the Transportation Accident impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Tsunami	A sea wave of local or distant origin that results from large-scale seafloor displacements associated with large earthquakes	Y	C3	The location of NMP2 along Lake Ontario precludes the possibility of a tsunami.

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Table E4-8 Evaluation of Other External Hazards				
Hazard	**Definition**	**Screened (Y/N)**	**Screening Criterion (Note a)**	**NINE MILE POINT 2 Response**
	or major submarine slides or landslides.			Based on this review, the Tsunami impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Turbine-generated missiles	The generation of a high-energy missile that is ejected from the turbine casing resulting from failure of a steam turbine. The turbine-generated missile may be ejected either upward (i.e., high-trajectory missile) which may result in damage to safety-related structures, systems, and components (SSCs) from the falling missile or it may be ejected directly toward safety-related SSCs (i.e., low-trajectory missiles).	Y	C2 PS2 PS4	Per USAR Section 3.5.1.3, at Unit 2, the original built-up type rotor design has been replaced with a monoblock type rotor design which reduces the susceptibility to stress corrosion cracking. Due to this replacement, the probability of missile generation from the Unit 2 turbine is statistically insignificant. Based on this review, the Turbine-Generated Missiles impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Volcanic activity	The extrusion of magma from beneath the earth's crust that may be accompanied by the flow of lava and explosion of fragmented material (pulverized pieces of rock, bits of chilled magma), and releases of volcanic ash and dust as well as gases and steam.	Y	C3	Not applicable to the site because of location (no active or dormant volcanoes located near plant site). Based on this review, the Volcanic Activity impact hazard can be considered to be negligible.

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Table E4-8
Evaluation of Other External Hazards

Hazard	Definition	Screened (Y/N)	Screening Criterion (Note a)	NINE MILE POINT 2 Response
				There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Waves	An area of moving water that is raised above the main surface of an ocean, a lake, etc. as a result of the wind blowing over an area of fluid surface.	Y	C3 C4	Waves associated with adjacent large bodies of water are not applicable to the site. Waves associated with external flooding are covered under that hazard. Based on this review, the Waves impact hazard can be considered to be negligible. There are no configuration-specific considerations for this hazard. This hazard can be excluded from the RICT program evaluation.
Note a: Refer to Table E4-9 for descriptions of the screening criteria				

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Table E4-9: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is < events for which plant is designed.	NUREG/CR-2300 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/y 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/y.	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	

7 Conclusions

Based on this analysis of external hazards for NMP2, no additional external hazards other than seismic events need to be added to the existing PRA model. The evaluation concluded that the hazards either do not present a design-basis challenge to NMP2, 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 real time risk (RTR) tool 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 seismic bounding delta CDF/LERF values.

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

8 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, October 12, 2012 (ADAMS Accession No. ML 12286A322).

2. Letter from Jennifer M. Golder (NRC) to Biff Bradley (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," 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 RAS-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, March 2017.

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

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

7. Individual Plant Examination of External Events (IPEEE) Submittal, Nine Mile Station – Unit 2, SAS-TR-95-001, June 1995.

8. Electric Power Research Institute (EPRI) NP-6041-SL, "A Methodology for Assessment of Nuclear Plant Seismic Margin," Revision 1, August 1991.

9. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 2, July 2014.

10. USAR, Final Safety Analysis Report, Revision 23, in *NMP Unit 2 USAR*.

11. Seismic Hazard and Screening Report (CEUS Sites), "Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the NTTF Review of the Fukushima Dai-ichi Accident," Attachment 3, Nine Mile Point Nuclear Station, Units 1 and 2, March 31, 2014 (ML14099A196).

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

12. N2-MISC-011, Nine Mile Point 2, "Nine Mile Point 2 Seismic CDF-LERF Estimate for RICT," Revision 1, August 20, 2019.

13. USNRC memorandum, "Safety/Risk Assessment Results for Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," September 2, 2010 (ADAMS ML100270582).

14. NMP2 2016 Full Power Internal Events (FPIE) PRA Update, March 2017.

15. Electric Power Research Institute (EPRI) TR 3002000709, "Final Report, Seismic Probabilistic Risk Assessment Implementation Guide," December 2013.

16. NMP2 Level 2 HRA Calculator Database, "NMP2 L2 HRA".

17. N2-PRA-015, Revision 1, "Nine Mile Point Unit 2 Probabilistic Risk Assessment Level 2 PRA Analysis".

18. Nine Mile Point Unit 2 Updated Safety Analysis Report (USAR) Revision 20, October 2012.

19. NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," Revision 2, February 2007.

20. N2-MISC-009, "Tornado Missile Risk Assessment for Nine Mile Point Unit 2," Revision 0.

21. NRC Regulatory Issue Summary (RIS) 2015-06, "Tornado Missile Protection," June 10, 2015.

22. NRC Enforcement Guidance Memorandum (EGM) 15-002, "Enforcement Discretion for Tornado-Generated Missile Protection Noncompliance," June 10, 2015.

23. Tornado Missile Protection Evaluation, Phase II, Nine Mile Point Unit 2, ECP-17-000411, December 12, 2017.

24. Nine Mile Point Nuclear Station, Units 1 & 2 Flood Hazard Reevaluation Report (FHRR), March 12, 2013, Docket Nos. 50-220 and 50-410.

25. Nine Mile Point Nuclear Station, Unit 2, Flooding Calculation WH-C-002, Rev. 0.

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

26. N2-PRA-001, Revision 2, “Nine Mile Point Unit 2 Probabilistic Risk Assessment Initiating Events Notebook”.

27. Not used

28. Not used

29. Not used

30. USNRC Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," LWR Edition, Revision 3, November 1978.

31. USNRC Standard Review Plan, "NUREG-0800 Chapter 3.5.1.6," Aircraft Hazards, Revision 4, March 2010.

32. USNRC Standard Review Plan, NUREG-0800, Section 2.2.3, "Evaluation of Potential Accidents," Revision 3, March 2007.

33. ER-AA-340, "GL 89-13 Program Implementing Procedure," Revision 9.

34. Nine Mile Point Nuclear Station, Unit 2, Technical Specifications Amendment 119.

ENCLOSURE 5

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

<u>Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)</u>

Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)

1. Introduction

Section 4.0, Item 6 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) requires that the license amendment request (LAR) provide the plant-specific total CDF and LERF to confirm applicability of the limits of Regulatory Guide (RG) 1.174, Revision 1 (Reference 3). (Note that RG 1.174, Revision 2 (Reference 4), issued by the NRC in May 2011, did not revise these limits.)

The purpose of this enclosure is to demonstrate that the Nine Mile Point Nuclear Station Unit 2 (NMP2) total Core Damage Frequency (CDF) and total Large Early Release Frequency (LERF) are below the guidelines established in RG 1.174. RG 1.174 does not establish firm limits for total CDF and LERF, but it 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 limits are met confirms that the risk metrics of NEI 06-09 can be applied to the NMP2 Risk Informed Completion Time (RICT) Program.

2. Technical Approach

Table E5-1 lists the NMP2 CDF and LERF point estimate values that resulted from a quantification of the baseline internal events (including internal flooding) and fire Probabilistic Risk Assessment (PRA) models (References 5 and 6, respectively). This table also includes an estimate of the seismic contribution to CDF and LERF based on the methodology detailed in Enclosure 4, Section 3. Other external hazards are below accepted screening criteria and therefore do not contribute significantly to the totals.

Table E5-1
Total Baseline CDF/LERF

NMP2 Baseline CDF			**NMP2 Baseline LERF**	
Source	**Contribution**		**Source**	**Contribution**
Internal Events PRA	1.8E-06		Internal Events PRA	2.6E-07
Fire PRA	3.1E-05		Fire PRA	6.2E-06
Seismic	6.4E-07		Seismic	3.2E-07
Other External Events	No significant contribution		Other External Events	No significant contribution
Total Unit 2 CDF	3.3E-05		**Total Unit 2 LERF**	6.8E-06

As demonstrated in Tables E5-1, 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, NMP2 TSTF-505 implementation is consistent with NEI 06-09 guidance. There will be a proceduralized check of the overall PRA results against the Reg Guide 1.174 thresholds in the PRA model update procedures.

Baseline Core Damage Frequency (CDF) and
Large Early Release Frequency (LERF)

3. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (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. Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, dated October 12, 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.

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 (Accession No. ML10091006).

5. N2-PRA-013 Revision 2, Summary Document Notebook, March 2019.

6. N2-PRA-021.11 Revision 1, Summary & Quantification (FQ) Notebook, July 2019.

ENCLOSURE 6

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

This enclosure is not applicable to the NMP2 submittal. Exelon is proposing to apply the Risk-Informed Completion Time Program only in Modes 1 and 2 and not in the shutdown Modes.

ENCLOSURE 7

License Amendment Request

Nine Mile Point Generating Station, Unit 2
Docket No. 50-410

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. Introduction

Section 4.0, Item 8 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) requires that the license amendment request (LAR) provide a discussion of the licensee's programs and procedures which assure the PRA models which support the RMTS are maintained consistent with the as-built/as-operated plant.

This enclosure describes the administrative controls and procedural processes applicable to the configuration control of PRA models used to support the Risk-Informed Completion Time (RICT) Program, which will be in place to ensure that these models reflect the as-built/as-operated plant. Plant changes, including physical modifications and procedure revisions, will be identified and reviewed prior to implementation to determine if they could impact the PRA models per ER- AA-600-1015, FPIE [Full Power Internal Events] PRA Model Update (Reference 3), and ER-AA-600-1061, Fire PRA Model Update and Control (Reference 4). The configuration control program will ensure these plant changes are incorporated into the PRA models as appropriate. The process will include discovered conditions associated with the PRA models, which will be addressed by the applicable site Corrective Action Program.

Should a plant change or a discovered condition be identified that has a significant impact to the RICT Program calculations as defined by the above procedures, an unscheduled update of the PRA model will be implemented. Otherwise, the PRA model change is incorporated into a subsequent periodic model update. Such pending changes are considered when evaluating other changes until they are fully implemented into the PRA models. Periodic updates are typically performed every two refueling cycles.

2. PRA Model Update Process

<u>Internal Event, Internal Flood, and Fire PRA Model Maintenance and Update</u>

The Fleet risk management process ensures that the applicable PRA model used for the RICT Program reflects the as-built/as-operated plant for Nine Mile Point Unit 2. The PRA configuration control process delineates the responsibilities and guidelines for updating the full power internal events, internal flood, and fire PRA models, and includes both periodic and unscheduled PRA model updates.

The process includes provisions for monitoring potential impact areas affecting 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), assessing the individual and cumulative risk impact of unincorporated changes, and controlling the model and necessary computer files, including those associated with the real time risk model.

Changes that are considered an upgrade per the ASME/ANS PRA standard receive a peer review focused on those aspects of the PRA model that represent the upgrade.

PRA Model Update Process

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 time risk model and the subsequent risk calculations which support the RICT Program (NEI 06-09, 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 guidance. Otherwise, the change is assigned a priority and is incorporated at a subsequent periodic update consistent with procedural requirements. (NEI 06-09, 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 (NEI 06-09, Section 2.3.4, Item 7.1, and 2.3.5, Item 9.1).

4. If a PRA model change is required for the real time risk model, but cannot be immediately implemented for a significant plant change or discovered condition, either:

    a. Interim analyses 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. The use of such bounding analyses is consistent with the guidance of NEI 06-09.

    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.

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

3. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (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. 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).

3. ER-AA-600-1015, "FPIE PRA Model Update."

4. ER-AA-600-1061, "Fire PRA Model Update and Control."

ENCLOSURE 8

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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 Risk Model

Attributes of the CRMP Model

1. Introduction

Section 4.0, Item 9 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) requires that the license amendment request (LAR) provide a description of PRA models and tools used to support the RMTS. This includes identification of how the baseline probabilistic risk assessment (PRA) model is 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 model and the CRMP tools, and training and qualification programs applicable to personnel responsible for development and use of the CRMP tools. NEI 06-09, Revision 0-A, uses the term CRMP for the program controlling the use of RMTS. This term is also used to designate the program implementing 10 CFR 50.65(a)(4) and the monitoring program for other risk informed LARs. To avoid confusion the term RICT program is used to indicate the program required by NEI 06-09, Revision 0-A, in lieu of the term CRMP. This item should also confirm that the RICT program 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 use in the real time risk (RTR) tool to support the Risk-Informed Completion Time (RICT) Program. The process employed to adapt the baseline models is demonstrated:

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

Quality controls and training programs applicable for the RICT Program are also discussed in this enclosure.

2. Translation of Baseline PRA Model for Use in Configuration Risk

The baseline PRA models for internal events, including internal flood and internal fire, are the peer-reviewed models. These models are updated when necessary to incorporate plant changes to reflect the as-built/as-operated plant. The internal flood model is integrated into the internal events model. These models will be used in the RICT Program. The models may be optimized for quantification speed but are verified to provide the same result as the baseline models in accordance with approved procedures.

The RTR tool will be used to facilitate all configuration-specific risk calculations and support the RICT Program implementation. The PRA Models utilize system initiator event fault trees so equipment unavailabilities are captured explicitly in these system initiator fault trees. Therefore, no adjustment to initiating event frequencies in required within the RTR tool.

Attributes of the CRMP Model

The baseline PRA models are modified as follows for use in configuration risk calculations:

- The unit availability factor is set to 1.0 (unit available).
- Maintenance unavailability is set to zero/false unless unavailable due to the configuration.
- Mutually exclusive combinations, including normally disallowed maintenance combinations, are adjusted to allow accurate analysis of the configuration.
- For systems where some trains are in service and some in standby, the RTR tool addresses the actual configuration of the plant including defining in service trains as needed.
- The impact of outside temperatures on system requirements like seasonal service water pumps were evaluated and found no dependent flags were needed to be addressed in the CRMP model.
- There are no changes in success criteria based on the time in the core operating cycle.

The configuration risk software is designed to quantify the unit-specific configuration for both internal events, including internal flooding and fire, and includes the seismic risk contribution when calculating the RMAT and RICT. Full quantifications will be used for each configuration. Pre-solved cutsets will be limited to results for specific configurations. For configurations without pre-solved cutsets the model will be quantified to produce cutsets for the previously unanalyzed configuration. If there are any changes in the underlying PRA, the PRA Results database in PARAGON will be updated in accordance with the RTR Update Procedure. The unique aspect of the configuration risk software for the RICT program is the quantification of fire risk and the inclusion of the seismic risk contribution. The other adjustments above are those used for the evaluation of risk under the 10CFR 50.65(a)(4) program.

The Nine Mile Point 2 (NMP2) PRA calculates Common Cause Basic Event (CCBE) probabilities from alpha factors and places the basic events under appropriate gates in the fault tree.

Adjustments to the CCF grouping or CCF probabilities are not necessary when a component is taken out-of-service for preventative maintenance:

- The component is not out-of-service for reasons subject to a potential common cause failure, and so the in-service components are not subject to increases in common cause probabilities.
- CCF relationships are retained for the remaining in-service components.
- The net failure probability for the in-service components includes the CCF contribution of the out-of-service component.

Attributes of the CRMP Model

As described in Reg Guide 1.177 (Reference 6), Section A-1.3.2.2, the CCF term should be treated differently when a component is taken down for preventive maintenance (PM) than as described for failure of a component. For PMs, the common cause factor is changed so that the model represents the unavailability of the remaining component. In the example provided in Reg Guide 1.177 for a 2-train system, the CCF event can be set to zero for PMs. This is done so that the model represents the unavailability of the remaining component, and not the common cause multiplier. The NMP2 approach is conservative in that for a 2-train system, the CCF event is retained for the component removed from service. Likewise, for systems with three or more trains, the CCF events that are related to the out-of-service component are retained.

The Vogtle RICT Safety Evaluation (Reference 5) describes the Vogtle approach for modeling common cause events with planned inoperability: "For planned inoperability, the licensee sets the appropriate independent failure to 'true' and makes no other changes while calculating a RICT." The NMP2 approach is the same as this Vogtle approach.

It is recognized that other modifications could be made to CCF factors for planned maintenance, particularly for common cause groups of three or more components. For example, in the Safety Evaluation (SE) in the Vogtle RICT Amendment (Reference 5), the NRC identifies a possible planned maintenance CCF modification to "modify all the remaining basic event probabilities to reflect the reduced number of redundant components."

Like Vogtle, the NMP2 CCF approach is a straightforward simplification that has inherent uncertainties. In the context of modifying CCF basic events for PMs, the Vogtle SE states the following:

> "The NRC staff also notes that common cause failure probability estimates are very uncertain and retaining precision in calculations using these probabilities will not necessarily improve the accuracy of the results. Therefore, the NRC staff concludes that the licensee's method is acceptable because it does not systematically and purposefully produce non-conservative results and because the calculations reasonably include common cause failures consistent with the accuracy of the estimates." (Reference 5)

The NMP2 approach for CCF during PMs is the same as the Vogtle approach; therefore, the NMP2 CCF approach is acceptable for RICT calculations, and adjusting the common cause grouping is not necessary for PMs. However, if a numeric adjustment is performed, the RICT calculation shall be adjusted to numerically account for the increased possibility of CCF in accordance with RG 1.177, as specified in Section A-1.3.2.1 of Appendix A of the RG.

For emergent conditions where the extent of condition is not completed prior to entering into the Risk Management Action Times or the extent of condition cannot rule out the potential for common cause failure, common cause RMAs are expected to be implemented to mitigate common cause failure potential and impact, in accordance with Exelon procedures. This is in line with the guidance of NEI 06-09 and precludes the need to adjust CCF probabilities. However, if a numeric adjustment is performed, the RICT calculation shall be adjusted to numerically account for the increased possibility of CCF in accordance with RG 1.177, as specified in Section A-1.3.2.1 of Appendix A of the RG.

Attributes of the CRMP Model

3. Quality Requirements and Consistency of PRA Model and Configuration Risk Tools

The approach for establishing and maintaining the quality of the PRA models, including the configuration risk 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 site's internal event, internal flood, and internal fire PRA models reasonably conform to the associated industry standards endorsed by Regulatory Guide 1.200 (Reference 3). This information provides a robust basis for concluding that the PRA models are of sufficient quality for use in risk-informed licensing actions.

For maintenance of an existing configuration risk model, changes made to the baseline PRA model in translation to the configuration risk model will be controlled and documented. Every PRA MOR Update results in an update to the RTR model in accordance with the FPIE and Fire PRA Update procedures. An acceptance test is performed after every configuration risk model update. This testing also verifies correct mapping of plant components to the basic events in the configuration risk model. The RTR model documentation includes changes made to the MOR model files to work with the RTR model software (e.g., quantification settings) along with verification that results are consistent between the RTR and PRA zero maintenance results. In addition, the RTR update for the MOR includes quantifying the RTR model for representative maintenance configurations and examining the results for appropriateness. These actions are procedurally controlled.

4. Training and Qualification

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

5. Application of the Configuration Risk Tool to the RICT Program Scope

The PARAGON software will be used to facilitate all configuration-specific risk calculations and support the RICT Program implementation. NMP2 has its own PARAGON model. This program is specifically designed to support implementation of RMTS. PARAGON will permit the user to evaluate all plant configurations using appropriate mapping of equipment to PRA basic events. The equipment in the scope of the RICT program will be able to be evaluated in the appropriate PRA models. The RICT program will meet RG 1.174 (Reference 4) and Exelon software quality assurance requirements.

Attributes of the CRMP Model

6. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (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. 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).

3. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March2009.

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.

5. Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Regarding Implementation of Topical Report Nuclear Energy Institute NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specification (RMTS) Guidelines," Revision 0-A (CAC NOS. ME9555 and ME9556), ML15127a669.

6. Nuclear Regulatory Commission, Regulatory Guide 1.177, MAY 2011, Revision 1.

ENCLOSURE 9

License Amendment Request

Nine Mile Point Unit 2 Nuclear Station
Docket No. 50-410

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. 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. Topical Report NEI 06-09-A (Reference 1), Section 2.3.4, item 10 requires an evaluation to determine insights that will be used to develop risk management actions (RMAs) to address these uncertainties. The baseline internal events PRA and fire PRA (FPRA) models document assumptions and sources of uncertainty and these were reviewed during the model peer reviews. The approach taken is, therefore, to review these documents to identify the items which 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 epistemic uncertainty analysis approach described below applies to the internal events PRA and any epistemic uncertainty impacts that are unique to FPRA are also addressed. In addition, Topical Report NEI 06-09-A requires that the uncertainty be addressed in RICT Program Configuration Risk Management Program (CRMP), otherwise referred to as the Real-Time Risk (RTR), tools by consideration of the translation from the PRA model to the RTR model. The RTR model, also referred to as the PARAGON model, discussed in Enclosure 8 includes internal events, 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 RTR tool during RICT Program calculations.

2. Assessment of Internal Events PRA Epistemic Uncertainty Impacts

In order to identify key sources of uncertainty for RICT Program application, an evaluation of internal events baseline PRA model uncertainty was performed, based on the guidance in NUREG-1855, rev.1 (Reference 2) and Electric Power Research Institute (EPRI) report TR-1016737 (Reference 3). 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 Nine Mile Point Unit 2 Nuclear Station (NMP2) baseline PRA model quantification (Reference 4) and the Fire PRA uncertainty evaluation (Reference 6).

Modeling uncertainties are considered in both the base PRA and in specific risk-informed applications. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the NMP2 internal events PRA technical elements are noted in the individual notebooks. These assumptions were collected from each notebook. The internal events PRA model uncertainties evaluation is documented in Reference 5 and considers the modeling uncertainties for the base PRA by identifying assumptions, determining if those assumptions are related to a source of modeling uncertainty and characterizing that uncertainty, as necessary. EPRI compiled a listing of generic sources of modeling uncertainty to be considered for each PRA technical element (Reference 3), and the evaluation performed for NMP2 (Reference 13) considered each of the generic sources of modeling uncertainty as well as the plant-specific sources.

Key Assumptions and Sources of Uncertainty

Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application4. No specific issues of PRA completeness have been identified relative to the TSTF-505 application, based on the results of the internal events PRA and fire PRA peer reviews.

Based on following the methodology in EPRI TR-1016737 for a review of sources of uncertainty, the impact of potential sources of uncertainty on the RICT application is discussed in Table E9-1, which identifies those potential sources that may be key sources of uncertainty for the RICT program. Note that RMAs will be developed when appropriate using insights from the PRA model results specific to the configuration.

Table E9-1
Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
NUREG/CR-6890 is used to develop the prior distribution for the LOOP initiator frequency and incorporates four causal categories (plant centered, switchyard centered, grid related, and weather related). The priors utilize industry data for the plant centered, switchyard centered, and weather LOOP categories. A Bayesian with plant specific data from January 1, 2008 – December 31, 2013 is utilized to obtain a posterior plant specific LOOP frequency. Finally, the generic industry data in NUREG/CR-6890 for the failure to recovery probabilities are utilized directly for the applicable time frames in the model.	LCOs related to offsite AC sources have an effect on the RICT	This approach provides a best estimate assessment for the site. However, the TSTF-505 procedure will require appropriate risk management action (RMA) focus on human performance for RICT entry, e.g., including an operator briefing on the significant human actions in the PRA that are pertinent to the configuration. Refer to Enclosure 12 for additional discussion on RMAs.

Key Assumptions and Sources of Uncertainty

Table E9-1
Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
The consequential LOOP failure probabilities are derived consistent with NUREG-CR-6890 data and industry practice. Consequential LOOP probability is modeled and is an important contributor to risk.	LCOs related to offsite AC sources have an effect on the RICT	This approach provides a best estimate assessment for the site. However, the TSTF-505 procedure will require appropriate risk management action (RMA) focus on human performance for RICT entry, e.g., including an operator briefing on the significant human actions in the PRA that are pertinent to the. Refer to Enclosure 12 for additional discussion on RMAs.
Individual CCF groups per system are included in the model for the suppression pool suction strainers based on generic and strainer plugging failure data and generic alpha factor data. All BWRs have improved their suppression pool suction strainers to reduce the potential for plugging. However, there is not a consistent method for the treatment of suppression pool strainer performance.	LCOs for which ECCS systems are involved in the RICT	Because suction strainer failures impact all ECCS systems as a common-mode failure, any potential extended unavailability via RICT is not relevant. This item does not represent a key source of uncertainty for the RICT Application.

Key Assumptions and Sources of Uncertainty

Table E9-1
Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Although ECCS pumps are designed to operate under saturated conditions, uncontrolled venting as a cause of core damage is not explicitly modeled because and EOP/training has operators control containment pressure in a band and not vent in an uncontrolled way (assumed to be very unlikely that both uncontrolled vent occurs and continues up to complete loss of ECCS suction).	LCOs for which containment heat removal systems are involved in the RICT	Uncontrolled venting has recently been added to the NMP2 PRA model (HEP ZCVUV) so that this source of uncertainty is now directly modeled in the PRA. Therefore, this will be directly included in RICT calculations and does not need to be separately addressed with uncertainty assessments. Because ECCS pumps are designed to operate under saturated conditions, it may now be conservative to assume failure of ECCS pumps given uncontrolled venting. However, such conditions go beyond saturated conditions in that steam will be flashing and establishing pump reliability given such a step-change in conditions would be difficult to justify. Modeling is judged to best represent the as-operated plant including explicit procedures to control containment venting pressure.

Key Assumptions and Sources of Uncertainty

Table E9-1
Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
Generally, credit for operation of systems beyond their design-basis environment is not taken. Some examples include the following: 1. HELB: HELB in turbine building is assumed to fail BOP systems in this location due to environment. 2. ISLOCA: HELB and ISLOCA in reactor building are assumed to fail equipment based on local impacts and other systems in the building in the long term (uncertainty about impacts – insufficient analysis for un-isolated breaks). Modeling is judged to be slightly conservative for HELB and ISLOCA, reasonable to conservative for containment failure and realistic for room cooling. Going beyond design basis does not guarantee failure and would likely impact mission time if severe enough. Failure early requires a local extreme impact, which is included in the evaluations.	LCOs for which ECCS systems are involved in the RICT.	HELB and ISLOCA events impact BOP and ECCS equipment directly such that any potential extended unavailability via RICT is not relevant (unavailable systems would be failed by the phenomena regardless). Therefore, this item does not represent a key source of uncertainty for the RICT Application.

Key Assumptions and Sources of Uncertainty

Table E9-1
Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
The Internal flood analysis and initiating event frequencies for spray, flood, and major flood scenarios developed consistent with the EPRI methodology. The 2019 FPIE update to internal flood uses a pipe length approach per latest revision of EPRI TR-1013141. One of the most important, and uncertain, inputs to an internal flooding analysis is the frequency of floods of various magnitudes (e.g., small, large, catastrophic) from various sources (e.g., clean water, untreated water, salt water, etc.). EPRI has developed some data, but the NRC has not formally endorsed its use.	All LCOs are involved in the RICT.	Updated industry data is developed routinely where it is common practice to implement this new data into the model during the next scheduled PRA Update. Therefore, the NMP2 PRA model will incorporate the new pipe rupture frequencies. Once implemented, this will not be a key source of uncertainty for the NMP2 RICT Application.
Detailed ISLOCA analysis includes the relevant considerations listed in IE-C12 of the ASME/ANS PRA Standard. Pipe rupture probability given failure of two normally closed valves is explicitly included in the accident sequence model. The accident sequence response model accounts for CCF, but ISLOCA initiators (failure of check valve and MOV) due to passive leak failures do not include common cause. ISLOCA is often a significant contributor to LERF. One key input to the ISLOCA analysis are the assumptions related to common cause rupture of isolation valves between the RCS/RPV and low pressure piping. There is no consensus approach to the data or treatment of this issue. Additionally, given an overpressure condition in low pressure piping, there is uncertainty surrounding the failure mode of the piping.	LCOs for which ECCS systems are involved in the RICT.	ISLOCA initiating event frequency is implemented in the model for each path individually. Probability of pressure boundary rupture is included in the model. The approach for the ISLOCA frequency determination is considered an industry good practice and probably a consensus model given the numerous studies since WASH-1400. In addition, ISLOCA frequency does not play a significant role in the RICT application because impacted systems are modeled and IE frequency would impact base and RICT models similarly such that CDF and LERF risk deltas would be insensitive to changes in ISLOCA frequency.

Key Assumptions and Sources of Uncertainty

Table E9-1 Assessment of Internal Events PRA Epistemic Uncertainty Impacts		
Source of Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
ISLOCA CCF: Common cause failures are included in the detailed ISLOCA evaluation but modeling of ISLOCA may have alternative hypotheses. If 95% bound ISLOCA frequency is used, LERF can increase significantly (a factor of 2 or more).	LCOs for which ECCS systems are involved in the RICT.	Common cause failures are included in the detailed ISLOCA evaluation and are considered good industry practice. While some overall conservatism could exist, it does not play a significant role in the RICT application because impacted systems are modeled and IE frequency would impact base and RICT models similarly such that risk deltas would be insensitive to changes regardless of initially unavailable components within credited equipment.
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. NMP2 assumes that a single LPCI pump is adequate and there is no real evidence yet that this is not acceptable to prevent core melt.	LCOs for which ECCS systems are involved in the RICT.	A set of sensitivities has been performed for NMP2 and this uncertainty has a minimal effect on CDF and LERF and does not alter the time period results of RICT calculations.

Key Assumptions and Sources of Uncertainty

Table E9-1
Assessment of Internal Events PRA Epistemic Uncertainty Impacts

Source of Uncertainty and Assumptions	TS LCOs	Model Sensitivity and Disposition
ATWS modeling conservative: • RPV Overpressure with successful ARI is assumed to cause a LOCA that fails feedwater. • ATWS modeling of operators preventing MSIV closure is not credited. • Operators preventing HCTL without main condenser available not credited.	LCOs for which Reactivity Control are involved in the RICT.	While some overall conservatism exist, the change in RICT-related risk due to this uncertainty is negligible because ATWS events lead to high conditional failure probabilities regardless of initially unavailable components of credited equipment. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.
Containment integrity following a vessel rupture event (i.e., excessive LOCA) is not assured. There is model uncertainty regarding the subsequent treatment that increases the likelihood of LERF for this extremely rare event.	All LCOs are involved in the RICT.	The current model treatment results in addition of a constant value to the CDF and LERF results. There is no impact for any RICT calculations. Therefore, this does not represent a key source of uncertainty for the RICT application.

3. Assessment of Translation (RTR Model) Uncertainty Impacts

Incorporation of the baseline PRA models into the RTR model used for RICT Program calculations may introduce new sources of model uncertainty. Table E9-2 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 RTR model.

Key Assumptions and Sources of Uncertainty

Table E9-2 Assessment of Translation Uncertainty Impacts			
RTR Model Change and Assumptions	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 both internal and fire PRAs.	The model, if restructured, will be logically equivalent and produce results comparable to the baseline 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. A conservative value for the seismic delta CDF is applicable.	Calculation of RICT and RMAT within RTR.	The addition of bounding impacts for seismic events has no impact on baseline PRA or RTR model. Impact is reflected in calculation of all RICTs and RMATs.	Since this is a bounding approach for addressing seismic risk in the RICT Program, it is not a source of translation uncertainty, and RICT Program calculations are not impacted, so no mandatory RMAs are required.
Set plant availability (Reactor Critical Years Factor) basic event to 1.0.	Typecode @AVAIL	Since the RTR model evaluates specific configurations during at-power conditions, the use of a plant availability factor less than 1.0 is not appropriate. This change allows the RTR model to produce appropriate results for specific at-power configurations.	This change is consistent with RTR tool practice; therefore this change does not represent a source of uncertainty, and RICT program calculations are not impacted, so no mandatory RMAs are required.

Key Assumptions and Sources of Uncertainty

4. Assessment of Supplementary FPRA Epistemic Uncertainty Impacts

The purpose of the following discussion is to address the epistemic uncertainty in the NMP2 FPRA. The NMP2 FPRA model includes various sources of uncertainty that exist because there is both inherent randomness in elements that comprise the FPRA and because the state of knowledge in these elements continues to evolve. The development of the NMP2 FPRA was guided by NUREG/CR-6850 (Reference 7). The NMP2 FPRA model used consensus models described in NUREG/CR-6850.

NMP2 used guidance provided in NUREG/CR-6850 and NUREG-1855 (Reference 2) to address uncertainties associated with FPRA for the RICT Program application. As stated in Section 1.3 of NUREG-1855:

> "Although the guidance in this report does not currently address all sources of uncertainty, the guidance provided on the uncertainty identification and characterization process and on the process of factoring the results into the decision making is generic and independent of the specific source of uncertainty. Consequently, the guidance is applicable for sources of uncertainty in PRAs that address at-power and low power and shutdown operating conditions, and both internal and external hazards."

NUREG-1855 also describes an approach for addressing sources of model uncertainty and related assumptions. It defines:

> "A source of model uncertainty exists when (1) a credible assumption (decision or judgment) is made regarding the choice of the data, approach, or model used to address an issue because there is no consensus and (2) the choice of alternative data, approaches or models is known to have an impact on the PRA model and results. An impact on the PRA model could include the introduction of a new basic event, changes to basic event probabilities, change in success criteria, or introduction of a new initiating event. A credible assumption is one submitted by relevant experts and which has a sound technical basis. Relevant experts include those individuals with explicit knowledge and experience for the given issue. An example of an assumption related to a source of model uncertainty is battery depletion time. In calculating the depletion time, the analyst may not have any data on the time required to shed loads and thus may assume (based on analyses) that the operator is able to shed certain electrical loads in a specified time."

NUREG-1855 defines consensus model as:

> "A model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group. In addition, widely accepted PRA practices may be regarded as consensus models. Examples of the latter include the use of the constant probability of failure on demand model for standby components and the Poisson model for initiating events. For risk-informed regulatory decisions, the consensus model approach is one that NRC has utilized or accepted for the specific risk-informed application for which it is proposed."

The potential sources of model uncertainty in the NMP2 FPRA model were evaluated for the 71 Fire PRA topics outlined in EPRI 1026511 (Reference 12). This guideline organizes the

Key Assumptions and Sources of Uncertainty

uncertainties in Topic Areas similar to those outlined in NUREG/CR-6850 and was used to evaluate the baseline FPRA epistemic uncertainty and evaluate the impact of this uncertainty on RICT Program calculations. Table E9-3 summarizes the results of the EPRI 1026511 review within the Topic Areas outlined by NUREG/CR-6850.

As noted above, the NMP2 FPRA was developed using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by NRC. Further, appropriate cable impacts were identified for the systems modeled in the Internal Events PRA and were modeled in the Fire PRA. No systems were conservatively assumed to be failed for all FPRA scenarios. Fire PRA methods were based on NUREG/CR-6850, other more recent NUREGs, e.g., NUREG-7150 (Reference 8), and published “frequently asked questions” (FAQs) for the FPRA.

It has been concluded that the uncertainties outlined in EPRI 1026511 do not present a significant impact on the NMP2 RICT calculations. Note that RMAs will be developed when appropriate using insights from the FPRA model results specific to the configuration.

Key Assumptions and Sources of Uncertainty

Table E9-3: Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition for RICT Application
1	Analysis boundary and partitioning	This task establishes the overall spatial scope of the analysis and provides a framework for organizing the data for the analysis. The partitioning features credited are required to satisfy established industry standards.	Based on the discussion of sources of uncertainly it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
2	Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the FPRA.	In the context of the FPRA, the uncertainty that is unique to the analysis is related to initiating event identification. However, that impact is minimized through use of the BWROG Generic Multiple Spurious Operation (MSO) list and the process used to identify and assess potential MSOs. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Component Selection task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

Table E9-3: Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition for RICT Application
3	Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. The overall process is essentially the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	Based on the discussion of sources of uncertainty it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
4	Qualitative Screening	Qualitative screening was performed; however, some structures (locations) were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the FPRA (based on industry guidance and criteria) were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip. However, such locations would not contain any features (equipment or cables identified in the prior two tasks) and consequently are expected to have a low risk contribution.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a CCDP commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

Table E9-3: Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
5	Fire-Induced Risk Model	The internal events PRA model was updated to add fire specific initiating event structure as well as additional system logic. The methodology used is consistent with that used for the internal events PRA model development and was subjected to industry Peer Review. The developed model is applied in such a fashion that all postulated fires are assumed to generate a plant trip. This represents a source of uncertainty, as it is not necessarily clear that fires would result in a trip. In the event the fire results in damage to cables and/or equipment identified in Task 2, the PRA model includes structure to translate them into the appropriate induced initiator.	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the FPRA development process would have reviewed significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

Table E9-3: Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition for RICT Application
6	Fire Ignition Frequency	Fire ignition frequency is an area with inherent uncertainty. Part of this uncertainty arises due to the counting and related partitioning methodology. However, the resulting frequency is not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the industry generic frequency values used for the FPRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates. NMP2 uses the ignition frequencies in NUREG-2169 (Reference 9) along with the revised heat release rates from NUREG - 2178 (Reference 10).	Based on the discussion of sources of uncertainty, it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would affect the RICT calculation. Consensus approaches are employed in the model. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
7	Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	The NMP2 FPRA did not screen out any fire scenarios based on low CDF/LERF contribution. That is, quantified fire scenarios results are retained in the cumulative CDF/LERF. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

Table E9-3: Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition for RICT Application
8	Scoping Fire Modeling	The framework of NUREG/CR-6850 includes two tasks related to fire scenario development. These two tasks are 8 and 11. The discussion of uncertainty for both tasks is provided in the discussion for Task 11.	See Task 11 discussion.
9	Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the FPRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. Hot short probabilities and hot short duration probabilities as defined in NUREG-7150, Volume 2 (Reference 8), based on actual fire test data, were used in the NMP2 Fire PRA. The uncertainty (conservatism) which may remain in the FPRA is associated with scenarios that do not contribute significantly to the overall fire risk. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

Table E9-3: Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
10	Circuit Failure Mode Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability and a hot short duration probability are assigned using industry guidance published in NUREG-7150, Volume 2 (Reference 8). The uncertainty values specified in NUREG-7150, Volume 2 are based on fire test data.	The use of hot short failure probability and duration probability is based on fire test data and associated consensus methodology published in NUREG-7150, Volume 2. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
11	Detailed Fire Modeling	The application of fire modeling technology is used in the FPRA to translate a fire initiating event into a set of consequences (fire induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and response of plant staff (detection, fire control, fire suppression). The fire modeling methodology itself is largely empirical in some respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a function of distance from the fire) are characterized as having some distribution	Consensus modeling approach is used for the Detailed Fire Modeling. The methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no additional RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

Table E9-3: Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
		(aleatory uncertainty). The epistemic uncertainty arises from the selection of the input parameters (specifically the HRR and growth rate) and how the parameters are related to the fire initiating event. While industry guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of that guidance retains that conservatism. The resulting risk estimates are also conservative.	

Key Assumptions and Sources of Uncertainty

Table E9-3: Fire PRA Sources of Model Uncertainty			
Task #	Description	Sources of Uncertainty	Disposition for RICT Application
12	Post-Fire Human Reliability Analysis	The Human Error Probabilities (HEPs) used in the FPRA were adjusted to consider the additional challenges that may be present given a fire. The HEPs included the consideration of degradation or loss of necessary cues due to fire. Given the methodology used, the impact of any remaining uncertainties is expected to be small.	The HEPs include the consideration of degradation or loss of necessary cues due to fire. The fire risk importance measures indicate that the results are somewhat sensitive to HRA model and parameter values. The NMP2 FPRA model HRA is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Assuming no credit for operator response is not realistic. However, the TSTF-505 procedure will require appropriate risk management action (RMA) focus on human performance for RICT entry, e.g., including an operator briefing on the significant human actions in the PRA that are pertinent to the configuration. Refer to Enclosure 12 for additional discussion on RMAs.

Key Assumptions and Sources of Uncertainty

Table E9-3: Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
13	Seismic-Fire Interactions Assessment	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	The qualitative assessment of seismic induced fires should not be a source of model uncertainty as it is not expected to provide changes to the quantified FPRA model. A conservative seismic hazard penalty is already applied to all RICT calculations to account for seismic risk impact. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Seismic-Fire Interactions Assessment task does not introduce any epistemic uncertainties that affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
14	Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit.	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

Table E9-3: Fire PRA Sources of Model Uncertainty

Task #	Description	Sources of Uncertainty	Disposition for RICT Application
15	Uncertainty and Sensitivity Analyses	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty.	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Uncertainty and Sensitivity Analyses task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.
16	FPRA Documentation	This task does not introduce any new uncertainties to the fire risk.	This task does not introduce any new uncertainties to the fire risk as it outlines documentation requirements. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the FPRA documentation task does not introduce any epistemic uncertainties that would affect the RICT calculation. Therefore, RICT Program calculations are not impacted, and no RMAs are required to address this item.

Key Assumptions and Sources of Uncertainty

5. Assessment of Level 2 Epistemic Uncertainty Impacts

In order to evaluate key sources of uncertainty for RICT Program application, an evaluation of Level 2 internal events PRA model uncertainty was performed, based on the guidance in NUREG-1855 (Reference 2) and Electric Power Research Institute (EPRI) report 1026511 (Reference 12). As described in NUREG-1855, sources of uncertainty include “parametric” uncertainties, “modeling” uncertainties, and “completeness” (or scope and level of detail) uncertainties.

The potential sources of model uncertainty in the NMP2 FPRA model were evaluated for the 32 Level 2 PRA topics outlined in EPRI 1026511 (Reference 12).

It has been concluded that the Level 2 related uncertainties outlined in EPRI 1026511 do not present a significant impact on the NMP2 RICT calculations. However, this review has highlighted the need to address time frames with the primary containment de-inerted in the calculations. This is necessary because de-inerted conditions are a configuration highlighted in importance within the Level 2 PRA results. Also note that RMAs will be developed when appropriate using insights from the Level 2 PRA model results specific to the configuration.

Key Assumptions and Sources of Uncertainty

6. References

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4. N2-PRA-014, Revision 1, "Nine Mile Point Unit 2 Nuclear Station Quantification Notebook,” March 2017.

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12. Electric Power Research Institute (EPRI) Technical Report TR-1026511, "Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," December 2012.

13. N2-MISC-012, Revision 0, “Assessment of Key Assumptions and Sources of Uncertainty for the Nine Mile Point Unit 2 Nuclear Station PRA”.

ENCLOSURE 10

License Amendment Request

Nine Mile Point Nuclear Station, Unit 2
Docket No. 50-410

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

<u>Program Implementation</u>

Program Implementation

1. Introduction

Section 4.0, Item 11 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09 (Reference 2) requires that the license amendment request (LAR) provide a description of the implementing programs and procedures regarding the 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).

2. RICT Program and Procedures

Exelon will develop a program description and implementing procedures 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. More detailed 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 included in NEI 06-09. The program will be integrated with the online work control process. The work control process currently identifies the need to enter a LCO Action statement as part of the planning process and will additionally identify whether the provisions of the RICT program are required for the planned work. The risk thresholds associated with 10CFR50.65(a)(4) will be coordinated with the RICT limits. The maintenance rule 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 TS and will be responsible for implementation of 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:

- 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.
- Limitations on implementing RICTs under voluntary and emergent conditions.
- Implementation of the RICT Program 30-day back stop limit.
- Use of the Real-Time Risk tool.

Program Implementation

- Guidance on recalculating RICT and risk management action time (RMAT) within 12 hours or within the most limiting front-stop CT after a plant configuration change.
- Requirements 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.
- 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. RICT Program Training

The scope of training for the RICT Program will include rules for the new TS program, Real-Time Risk tool software, TS Actions included in the program, and procedures. This training will be conducted for the following Exelon personnel:

Site Personnel

- Operations Director
- Operations Personnel (Licensed and Non-Licensed)
- Operations Training
- Outage Manager
- On-line Manager
- Planning and Scheduling Personnel
- Work Week Managers
- Regulatory Assurance Personnel
- Selected Maintenance Personnel
- Engineering
- Risk Management
- Other Selected Management

Corporate Personnel

- Operations Corporate Functional Area Manager
- Fleet Outages Corporate Functional Area Manager
- Licensing Management and Personnel
- Risk Management Personnel and Managers
- Training Management and Personnel
- Other Selected Management

Training will be carried out in accordance with Exelon training procedures and processes. These procedures were written based on the Institute of Nuclear Power Operations (INPO) Accreditation (ACAD) requirements, as developed and maintained by the National Academy for Nuclear Training. Exelon has planned three levels of training for implementation of the RICT Program. They are described below:

Program Implementation

Level 1 Training

This is the most detailed training. It is intended for the individuals who will be directly involved in the implementation of the RICT Program. This level of training includes the following attributes:

- Specific training on the revised TS
- Record keeping requirements
- Case studies
- Hands-on experience with the Real-Time Risk tool for calculating RMAT and RICT
- Identifying appropriate RMAs
- Common cause failure RMA considerations in emergent RICTs
- Other detailed aspects of the RICT Program

Level 2 Training

This training is applicable to plant 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. These individuals need a broad understanding of the purpose, concepts, and limitations of the RICT Program. Level 2 training is significantly more detailed than Level 3 training (described below), but it is different from Level 1 training in that hands-on time with the Real-Time Risk tool, case studies, and other specifics are not required.

Level 3 Training

This training is intended for the remaining personnel who require an awareness of the RICT Program. These employees need basic knowledge of RICT Program requirements and procedures. This training will cover RICT Program concepts that are important to disseminate throughout the organization.

4. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (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. 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).

ENCLOSURE 11

License Amendment Request

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Monitoring Program

Monitoring Program

1. Introduction

Section 4.0, Item 12 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09 (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 (Reference 2). (Note that RG 1.174, Revision 2 (Reference 4), issued by the NRC in May 2011, made editorial changes to the applicable section referenced in the NRC safety evaluation for Section 4.0, Item 12.)

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, Risk Informed Technical Specifications Initiative 4b (Reference 2). General requirements for a Performance Monitoring Program for risk-informed applications are discussed in Element 3 of Regulatory Guide (RG) 1.174 (Reference 3).

2. 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 (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), and the total risk 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.

The total average annual change in risk for extended CTs will be compared to the guidance of RG 1.174, Figures 4 and 5 (Reference 4), 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 ensure future plant operation is within the guidelines will be conducted under the corrective action program.

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

- RICT applications that dominated the risk increase
- Risk contributions from planned vs. emergent RICT applications
- Risk Management Actions (RMA) implemented but not credited in the risk calculations
- Risk impact from applying RICT to avoid multiple shorter duration outages
- Any specific RICT application that incurred a large proportion of the risk

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

- Administrative restrictions on 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 (SSC)
- Plant modifications to reduce risk impact of future planned maintenance configurations

In addition to impacting cumulative risk, implementation of the RICT Program may potentially impact the unavailability of SSCs. The existing Maintenance Rule (MR) monitoring programs under 10 CFR 50.65(a)(1) and (a)(2) provide for evaluation and disposition of unavailability impacts which may be incurred from implementation of the RICT Program. The SSCs in the scope of the RICT Program which are also in the scope of the MR allows the use of the MR Program.

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

3. References

1. Letter from Jennifer M. Golder (NRC) to Biff Bradley (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. 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, 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.

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.

ENCLOSURE 12

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Risk Management Action Examples

Risk Management Action Examples

1. 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 RMAs. RMAs will be governed by plant procedures for planning and scheduling maintenance activities. 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, Revision 0-A (Reference 1).

2. Responsibilities

For planned entries into the RICT Program, Work Management is responsible for developing the RMAs with assistance from Operations and Risk Management. Operations is responsible for approval and implementation of RMAs. For emergent entry into extended CTs, Operations is also responsible for developing the RMAs.

3. 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 a RMAT already in place, the procedure will require a re-evaluation of the existing RMAs for the new plant configuration to determine if new 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 SSCs and reducing the likelihood of initiating events relying on the affected function will be developed to address the increased likelihood of a common cause event.

RMAs will be implemented in accordance with current procedures (e.g., References 2, 3, 4, 5) no later than the time at which an incremental core damage probability (ICDP) of 1E-6 is reached, or no later than the time when an incremental large early release probability (ILERP) of 1E-7 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-3 per year or 1E-4 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 structures, systems, or components (SSCs) are most important from a CDF or LERF perspective for a specific plant configuration, RMAs may be created to protect these SSCs. Similarly, knowledge of the initiating event or sequence contribution 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 6) and EPRI TR-1026511 (Reference 7) 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, or important support systems or human actions). Enclosure 9 identifies several areas of uncertainty in the internal events and fire PRAs that will be considered in defining configuration-specific RMAs when entering a RICT.

Risk Management Action Examples

If the planned activity or emergent condition includes a SSC that is identified to impact Fire PRA, as identified in the current Real-Time Risk Program, Fire PRA specific RMAs associated with that SSC will be implemented per the current plant procedure.

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 to determine RICTs, the 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.

   - Shift brief
   - Pre-job brief
   - Training
   - Presence of system engineer or other expertise related to the activity
   - Special purpose procedure to identify risk sources and contingency plans

2) Actions to reduce the duration of maintenance activities.

   - Pre-staging materials
   - Conducting training on mock-ups
   - Performing the activity around the clock
   - Performing walk-downs on the actual system(s) to be worked on prior to beginning work

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

   - Suspend or minimize activities on 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 meant to mitigate
   - Use temporary equipment to provide backup power, ventilation, etc.
   - Reschedule other risk-significant activities

Determination of RMAs involves the use of both qualitative and quantitative considerations for the specific plant configuration and the practical means available to manage risk. The scope and number of RMAs developed and implemented are reached in a graded manner.

Procedural guidance for development of RMAs in support of the RICT program builds off the RMAs developed for other processes, such as the RMAs developed under the 10CFR 50.65(a)(4) program and the protected equipment program. Additionally, Common Cause RMAs are developed to address the potential impact of common cause failures.

General RMAs are developed for input into the RICT system guidelines. These guidelines are listed in site-specific T&RMs and are developed using a graded approach. Consideration is given for system functionality and includes consideration for common cause impacts within the system. These RMAs include:

- 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 maintenance

In addition to the RMAs developed qualitatively for the system guidelines, RMAs are developed based on the Real-Time Risk tool 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 trains for protection
- Identification of important Operator Actions for briefings
- Identification of key fire initiators and fire zones for RMAs in accordance with the site Fire RMA process
- Identification of dominant initiating events and actions to minimize potential for initiators
- Consideration of insights from PRA model cutsets, through comparison of importances

Common cause RMAs are also developed to ensure availability of redundant SSCs, to ensure availability of diverse or alternate systems, to reduce the likelihood of initiating events that require operation of the out-of-service components, 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 trains
- Confidence runs performed for standby SSCs
- Increased monitoring for running components
- Expansion of monitoring for running components
- Deferring maintenance and testing activities that could generate 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 which focus on initiating event response or loss of potentially affected SSCs

Per Exelon procedure, for emergent conditions where the extent of condition is not performed prior to entering into the Risk Management Action Times or the extent of condition cannot rule out the potential for common cause failure, common cause RMAs are expected to be implemented to mitigate common cause failure potential and impact. These can include the pre-identified RMAs included in the system guidelines as discussed above, as well as alternative common cause RMAs for the specific configuration. Alternate RMAs, including both regular and common cause considerations, are developed for the specific configuration following the steps outlined above.

4. Examples

Representative examples of RMAs that may be considered during a RICT Program entry, to reduce the risk impact and ensure adequate defense-in-depth, for electrical equipment and for a low pressure Emergency Core Cooling System (ECCS) loop out of service are provided below.

4.1. Electrical Action Statements

Risk Management Action Examples

4.1.1. For TS 3.8.1.B, one required DG inoperable, additional RMAs would include:

1. Actions to increase risk awareness and control.

   - Briefing of the on-shift Operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a Loss of Offsite Power and station blackout including bus crossties.

   - Notification of the TSO of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred.

   - Proactive implementation of RMAs during times of high grid stress conditions, such as during high demand conditions.

2. Actions to reduce the duration of maintenance activities.

   - For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability.

   - Confirmation of parts availability prior to entry into a preplanned RICT.

   - Walkdown of work prior to execution.

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

   - Evaluation of weather conditions for threats to the reliability of offsite power supplies.

   - Deferral of elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and auxiliary transformers associated with the unit.

   - Deferral of planned maintenance or testing that affects the reliability of operable DGs and their associated support equipment. Treat the remaining operable DGs as protected equipment.

   - Deferral of planned maintenance or testing on redundant train safety systems. If testing or maintenance activities must be performed, a review of the potential risk impact will be performed.

   - Implementation of 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected DG.

4.1.2. For TS 3.8.1.A, one required offsite circuit inoperable, additional RMAs would include:

1. Actions to increase risk awareness and control.

Risk Management Action Examples

- Briefing of the on-shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a Loss of Offsite Power and station blackout.

- Notification of the TSO of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred.

- Proactive implementation of RMAs during times of high grid stress conditions prior to reaching the RMAT, such as during high demand conditions.

2. Actions to reduce the duration of maintenance activities.

    - For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability

    - Confirmation of parts availability prior to entry into a preplanned RICT.

    - Walkdown of work prior to execution.

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

    - Evaluation of weather conditions for threats to the reliability of remaining offsite power supplies.

    - Deferral of elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and station reserve transformers associated with the unit.

    - Protection of the remaining offsite source, including switchyard and transformer.

    - Deferral of planned maintenance or testing that affects the reliability of DGs and their associated support equipment. Treat these as protected equipment.

    - Implementation of 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected offsite source.

4.1.3. For TS 3.8.1.D, one required offsite circuit inoperable AND one required DG inoperable, additional RMAs would include:

1. Actions to increase risk awareness and control.

    - Briefing of the on-shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a Loss of Offsite Power and station blackout.

    - Notification of the TSO of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred.

Risk Management Action Examples

- Proactive implementation of RMAs during times of high grid stress conditions prior to reaching the RMAT, such as during high demand conditions.
- For a planned RICT, prior to removal from service the actions in the loss of bus procedure associated with the inoperable DG would be reviewed.

2. Actions to reduce the duration of maintenance activities.
   - For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability.
   - Confirmation of parts availability prior to entry into a preplanned RICT.
   - Walkdown of work prior to execution.
3. Actions to minimize the magnitude of the risk increase.
   - Deferral of elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and station reserve transformers associated with the unit.
   - Deferral of planned maintenance or testing that affects the reliability of DGs and their associated support equipment for the remaining buses.
   - Implementation of 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected bus.
   - Place unaffected trains of systems into service. For example, if one of two instrument nitrogen compressors is powered by the affected bus, the other unaffected compressor would be placed in service to support containment atmosphere control. This would be done prior to entry into a planned RICT.

4.1.4. TS 3.8.4.A, Division 1 or 2 DC electrical power subsystem inoperable, additional RMAs would include:

1. Actions to increase risk awareness and control.
   - Briefing of the on-shift operations crew concerning the unit activities, including any compensatory measures established, and review of the appropriate emergency operating procedures for a Loss of DC division and station blackout.
   - Briefing of the on-shift operations crew concerning the impact the DC division has on the potential response to plant events such as reduced control systems.
   - Prior to removal from service. If a planned RICT, the actions in the associated loss of bus procedure would be reviewed and implemented.
   - Minimize activities that could trip the unit.

Risk Management Action Examples

2. Actions to reduce the duration of maintenance activities.

   - For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability.
   - Confirmation of parts availability prior to entry into a preplanned RICT.
   - Walkdown of work prior to execution.

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

   - Deferral of elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and auxiliary transformers associated with the unit.
   - Protection of the remaining DC electrical buses.
   - Remove nonessential loads from battery to extend time voltage will remain above minimum required level.
   - Implementation of 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected bus.

4.2. ECCS Action Statements

4.2.1 For TS Action 3.5.1.A, one low pressure ECCS injection/spray subsystem inoperable, the RMAs would include the following:

1. Defer planned maintenance or testing activities on the redundant ECCS low pressure injection loops and associated support equipment. Treat those systems as protected equipment.
2. Defer planned maintenance or testing that affects the reliability of those safety systems that provide a defense-in-depth. If testing or maintenance activities must be performed, a review of the potential risk impact will be performed.
3. Minimize activities that could trip the unit.
4. Verify system alignment of remaining loops of low pressure ECCS.
5. Implement 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected ECCS loop.

5. 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).

2. Exelon Procedure OP-AA-201-012-1001, "Operations On-Line Fire Risk Management."

3. N2-CRM-003, "Development of Risk Management Actions for Inclusion of Fire Risk Insights to the Nine Mile Point Unit 2 Generating Station Risk Management Process."

4. Exelon Procedure WC-AA-101-1006, "On-Line Risk Management and Assessment."
5. Exelon Procedure OP-AA-108-117, "Protected Equipment Program."

6. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," U.S. Nuclear Regulatory Commission, Revision 1 March 2017.

7. EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," Technical Update, Electric Power Research Institute, December 2012.