UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 12, 2021

Mr. Thomas A. Conboy
Site Vice President
Northern States Power Company - Minnesota
Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT NO. 206 RE: TSTF-505, REVISION 2, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES – RITSTF INITIATIVE 4b" (EPID L-2020-LLA-0062)

Dear Mr. Conboy:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 206 to Renewed Facility Operating License No. DPR-22, for the Monticello Nuclear Generating Plant. The amendment consists of changes to the technical specifications (TSs) in response to your application dated March 30, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20090F820), as supplemented by letters dated December 21, 2020 (ADAMS Accession No. ML20356A131), April 20, 2021 (ADAMS Accession No. ML21110A666), and June 30, 2021 (ADAMS Accession No. ML21181A308).

The amendment revises TS requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation (LCOs) are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b."

A copy of the Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Robert F. Kuntz, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosure:
1. Amendment No. 206 to DPR-22
2. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 206
Renewed License No. DPR-22

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company dated March 30, 2020, as supplemented by letters dated December 21, 2020, April 20, 2021, and June 30, 2021, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-22 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 206, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 180 days of the date of issuance. The implementation of the amendment shall include the items listed in Table A5-1, "RICT Program PRA Implementation Items" of the Northern States Power Company letter dated April 20, 2021.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2021.07.12 15:24:14 -04'00'

Nancy L. Salgado, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: July 12, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 206

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Renewed Facility Operating License No. DPR-22

Replace the following page of the Renewed Facility Operating License No. DPR-22 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

INSERT	REMOVE
Page 3	Page 3

Technical Specifications

Replace the following pages of Appendix A, Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

INSERT	REMOVE	INSERT	REMOVE	INSERT	REMOVE
1.3-11	1.3-11	3.3.4.1-2	3.3.4.1-2	3.5.1-1	3.5.1-1
1.3-12	--------	3.3.4.1-3	3.3.4.1-3	3.5.1-2	3.5.1-2
3.1.7-1	3.1.7-1	3.3.4.1-4	--------	3.5.1-3	3.5.1-3
3.1.7-2	3.1.7-2	3.3.5.1-2	3.3.5.1-2	3.5.1-4	3.5.1-4
3.1.7-3	3.1.7-3	3.3.5.1-3	3.3.5.1-3	3.5.1-5	3.5.1-5
3.1.7-4	3.1.7-4	3.3.5.1-4	3.3.5.1-4	3.5.1-6	3.5.1-6
3.1.7-5	3.1.7-5	3.3.5.1-5	3.3.5.1-5	3.5.1-7	3.5.1-7
3.1.7-6	3.1.7-6	3.3.5.1-6	3.3.5.1-6	3.5.1-8	3.5.1-8
3.1.7-7	--------	3.3.5.1-7	3.3.5.1-7	3.5.1-9	--------
3.3.1.1-1	3.3.1.1-1	3.3.5.1-8	3.3.5.1-8	3.5.3-1	3.5.3-1
3.3.1.1-2	3.3.1.1-2	3.3.5.1-9	3.3.5.1-9	3.6.1.2-3	3.6.1.2-3
3.3.1.1-3	3.3.1.1-3	3.3.5.1-10	3.3.5.1-10	3.6.1.3-1	3.6.1.3-1
3.3.1.1-4	3.3.1.1-4	3.3.5.1-11	3.3.5.1-11	3.6.1.3-2	3.6.1.3-2
3.3.1.1-5	3.3.1.1-5	3.3.5.1-12	3.3.5.1-12	3.6.1.3-3	3.6.1.3-3
3.3.1.1-6	3.3.1.1-6	3.3.5.1-13	--------	3.6.1.3-4	3.6.1.3-4
3.3.1.1-7	3.3.1.1-7	3.3.5.1-14	--------	3.6.1.3-5	3.6.1.3-5
3.3.1.1-8	3.3.1.1-8	3.3.5.2-1	3.3.5.2-1	3.6.1.3-6	3.6.1.3-6
3.3.1.1-9	3.3.1.1-9	3.3.5.2-2	3.3.5.2-2	3.6.1.3-7	3.6.1.3-7
3.3.1.1-10	3.3.1.1-10	3.3.6.1-1	3.3.6.1-1	3.6.1.3-8	3.6.1.3-8
3.3.1.1-11	3.3.1.1-11	3.3.6.1-2	3.3.6.1-2	3.6.1.3-9	--------
3.3.1.1-12	--------	3.3.7.2-1	3.3.7.2-1	3.6.1.3-10	--------
3.3.2.2-1	3.3.2.2-1	3.3.7.2-2	3.3.7.2-2	3.6.1.6-1	3.6.1.6-1
3.3.2.2-2	3.3.2.2-2	3.3.7.2-3	3.3.7.2-3	3.6.1.6-2	3.6.1.6-2
3.3.2.2-3	3.3.2.2-3	3.3.8.1-1	3.3.8.1-1	3.6.1.6-3	--------
3.3.4.1-1	3.3.4.1-1	3.4.3-1	3.4.3-1	3.6.1.7-1	3.6.1.7-1

INSERT	REMOVE
3.6.1.8-1	3.6.1.8-1
3.6.1.8-2	3.6.1.8-2
3.6.2.3-1	3.6.2.3-1
3.6.2.3-2	3.6.2.3-2
3.7.1-1	3.7.1-1
3.7.1-2	3.7.1-2
3.7.2-1	3.7.2-1
3.7.2-2	3.7.2-2
3.7.2-3	---------
3.8.1-1	3.8.1-1
3.8.1-2	3.8.1-2
3.8.1-3	3.8.1-3
3.8.1-4	3.8.1-4
3.8.1-5	3.8.1-5
3.8.1-6	3.8.1-6
3.8.1-7	3.8.1-7
3.8.1-8	3.8.1-8
3.8.1-9	3.8.1-9
3.8.1-10	3.8.1-10
3.8.1-11	---------

2. Pursuant to the Act and 10 CFR Part 70, NSPM to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operations, as described in the Final Safety Analysis Report, as supplemented and amended, and the licensee's filings dated August 16, 1974 (those portions dealing with handling of reactor fuel);

3. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

4. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

5. Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess, but not separate, such byproduct and special nuclear material as may be produced by operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission, now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level

   NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 2004 megawatts (thermal).

2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 206, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. Physical Protection

   NSPM shall implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search

1.3 Completion Times

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 OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

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

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

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

1.3 Completion Times

EXAMPLES (continued)

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.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

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. Concentration of sodium pentaborate in solution not within limits of Figure 3.1.7-1 and Table 3.1.7-1 Equation 2, but available volume of sodium pentaborate solution is within limits of Table 3.1.7-1 Equation 1.	A.1 Restore concentration of sodium pentaborate in solution to within limits.	7 days
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1 Restore SLC subsystem to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1 Restore one SLC subsystem to OPERABLE status.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.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 or Equation 1 of Table 3.1.7-1.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is within the limits of Figure 3.1.7-2.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.3	Verify temperature of room in the vicinity of the SLC pumps is within the solution temperature limits of Figure 3.1.7-2 or verify SLC pump suction lines heat tracing is OPERABLE.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.4	Verify continuity of explosive charge.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.5	Verify the concentration of sodium pentaborate in solution is within the limits of Figure 3.1.7-1 or within the limits of Equation 2 of Table 3.1.7-1.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 24 hours after water or sodium pentaborate is added to solution <u>AND</u> Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2
SR 3.1.7.6	Verify each SLC subsystem manual valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 24 gpm at a discharge pressure ≥ 1275 psig.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	In accordance with the Surveillance Frequency Control Program <u>AND</u> --------NOTE------- Only required if SLC pump suction lines heat tracing is inoperable. ----------------------- Once within 24 hours after room temperature in the vicinity of the SLC pumps is restored within the solution temperature limits of Figure 3.1.7-2
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ 55.0 atom percent B-10.	Prior to addition to SLC tank

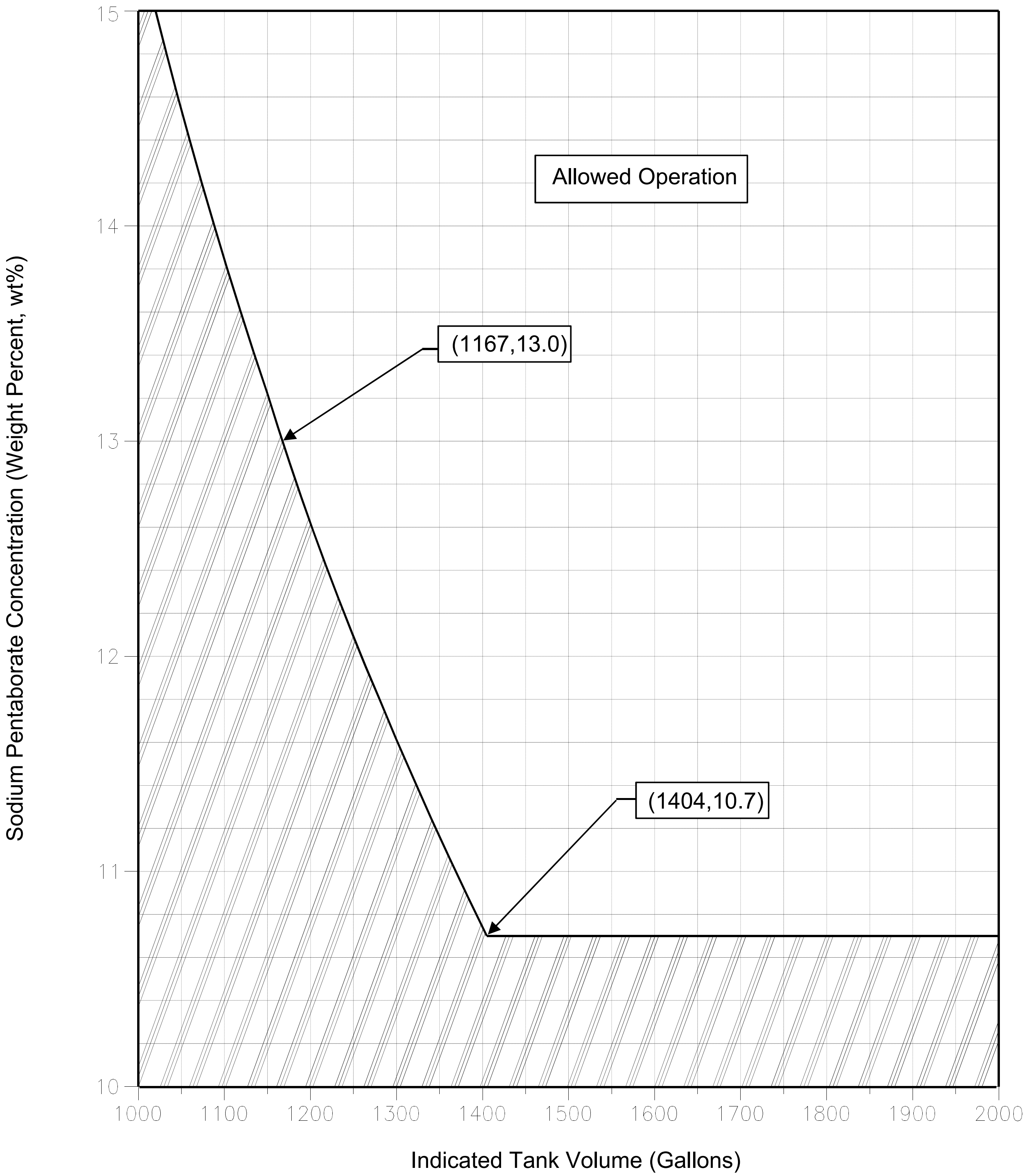

Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume Versus Concentration Requirements

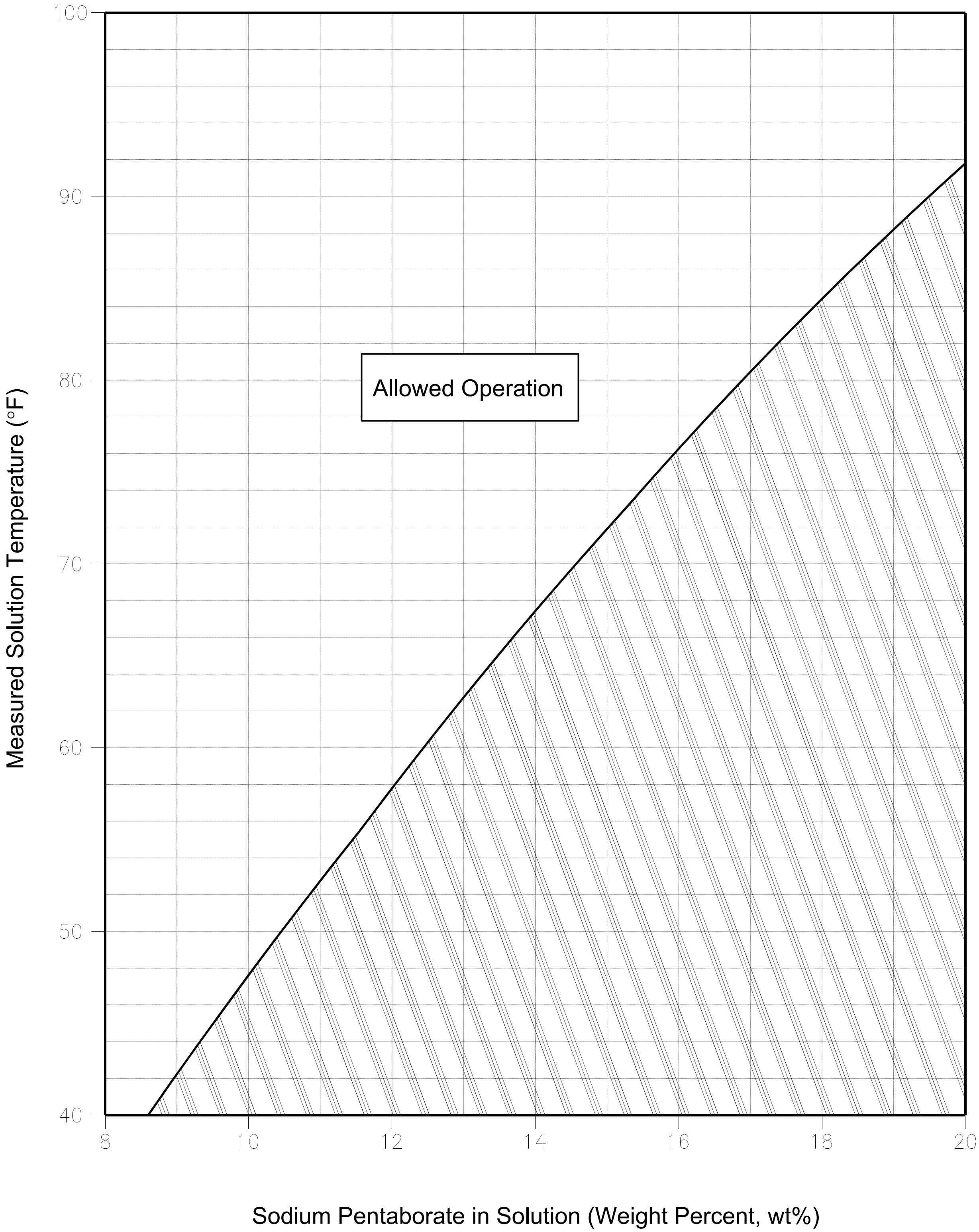

Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Solution Temperature Versus Concentration Requirements

Table 3.1.7-1 (page 1 of 1)
Equations for Required Sodium Pentaborate Tank Volume and Concentration

<u>Equation 1</u>

$$V \geq \left(\frac{71.18}{0.0051 \times C + 0.998}\right)\left(1+\frac{4821}{1101-E}\right)\left(\frac{19.8}{E}\right)\left(\frac{100}{C}\right)+128 \text{ gal}$$

Where:

C = measured boron solution concentration (wt%)
E = measured boron solution enrichment (atom%)
V = indicated boron solution tank volume (gal)

<u>Equation 2</u>

$$C \geq 8.28\left(\frac{86}{Q}\right)\left(\frac{19.8}{E}\right)$$

Where:

C = measured boron solution concentration (wt%)
E = measured boron solution enrichment (atom%)
Q = measured pump flow rate (gpm) at 1275 psig

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---

1. Separate Condition entry is allowed for each channel.

2. When the Function 2.b and 2.c channels are not within the limit of SR 3.3.1.1.2 due to APRM indication not within limits, entry into associated Conditions and Required Actions may be delayed for up to 2 hours if the APRM is indicating a lower power value than the calculated power, and for up to 12 hours if the APRM is indicating a higher power value than the calculated power.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. <u>OR</u>	12 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	--------------NOTE------------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, 2.f or 2.g. ------------------------------------ Place associated trip system in trip.	12 hours OR In accordance with the Risk Informed Completion Time Program
B. -----------NOTE------------ Not applicable for Functions 2.a, 2.b, 2.c, 2.d, 2.f or 2.g. ------------------------------- 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.	6 hours OR In accordance with the Risk Informed Completion Time Program
	OR		
	B.2	Place one trip system in trip.	6 hours OR In accordance with the Risk Informed Completion Time Program
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to ≤ 40% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
	<u>AND</u>		
	F.2	--------------NOTE------------- Only applicable to Function 5. ------------------------------------ Reduce reactor pressure to < 600 psig.	12 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours
	<u>AND</u>		
	I.2	-------------NOTE---------------- LCO 3.0.4 is not applicable ------------------------------------- Restore required channels to OPERABLE.	120 days
J. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	J.1	Reduce THERMAL POWER below the MELLLA boundary defined in the COLR.	12 hours
K. Required Action and associated Completion Time of Condition I or J not met.	K.1	Reduce THERMAL POWER to < 20% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

--NOTES--

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. 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 for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.2	--------------------------------NOTE------------------------------- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.3	--------------------------------NOTE------------------------------- Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.4	Perform a functional test of each RPS automatic scram contactor.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.5	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.6	Calibrate the local power range monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.7	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.8	Calibrate the trip units.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.9	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.10	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	-------------------------------NOTES------------------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. --- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.12	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.13	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low Functions are not bypassed when THERMAL POWER is > 40% RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.14	Verify the RPS RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.15	-------------------------------NOTES------------------------------ 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For Functions 2.b and 2.f, the CHANNEL FUNCTIONAL TEST includes the recirculation flow input processing, excluding the flow transmitters. --- Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.16	Verify the oscillation power range monitor (OPRM) function is not bypassed when APRM Simulated Thermal Power is ≥ 25% RTP and drive flow is ≤ 60% of rated drive flow.	In accordance with the Surveillance Frequency Control Program

Table 3.3.1.1-1 (page 1 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
	5[(a)]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
b. Inop.	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.12	NA
	5[(a)]	3	H	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.12	NA
2. Average Power Range Monitors					
a. Neutron Flux – High, (Setdown)	2	3[(c)]	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 20% RTP
b. Simulated Thermal Power – High	1	3[(c)]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 0.61W + 67.2% RTP[(b)] and ≤ 116% RTP

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ≤ 0.55 (W – Delta W) + 61.5% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." The cycle-specific value for Delta W is specified in the COLR.

(c) Each APRM / OPRM channel provides inputs to both trip systems.

Table 3.3.1.1-1 (page 2 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
c. Neutron Flux – High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11(f)(g) SR 3.3.1.1.15	≤ 122% RTP
d. Inop.	1, 2	3(c)	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
e. 2-Out-Of-4 Voter	1, 2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	NA
f. OPRM Upscale	≥ 20% RTP	3(c)	I	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.16	As specified in COLR
g. Extended Flow Window Stability – High	Within EFW boundary defined in COLR	3(c)	J	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11 SR 3.3.1.1.15	As specified in COLR
3. Reactor Vessel Steam Dome Pressure – High	1, 2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.14	≤ 1075 psig

(c) Each APRM / OPRM channel provides inputs to both trip systems.

(f) If the as-found channel setpoint is not the Nominal Trip Setpoint but is conservative with respect to the Allowable Value, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(g) The instrument channel setpoint shall be reset to the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The NTSP and the methodology used to determine the NTSP are specified in the Technical Requirements Manual.

Table 3.3.1.1-1 (page 3 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Reactor Vessel Water Level – Low	1, 2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.14	≥ 7 inches
5. Main Steam Isolation Valve – Closure	1, 2[d]	8	F	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure – High	1, 2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12	≤ 2.0 psig
7. Scram Discharge Volume Water Level – High					
a. Resistance Temperature Detector	1, 2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12	≤ 56.0 gallons
	5[a]	2	H	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12	≤ 56.0 gallons
b. Float Switch	1, 2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12	≤ 56.0 gallons
	5[a]	2	H	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12	≤ 56.0 gallons

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(d) With reactor pressure ≥ 600 psig.

Table 3.3.1.1-1 (page 4 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Turbine Stop Valve – Closure	> 40% RTP	4	E	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
9. Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure – Low	> 40% RTP	2	E	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 167.8 psig
10. Reactor Mode Switch – Shutdown Position	1, 2	1	G	SR 3.3.1.1.10 SR 3.3.1.1.12	NA
	5[(a)]	1	H	SR 3.3.1.1.10 SR 3.3.1.1.12	NA
11. Manual Scram	1, 2	1	G	SR 3.3.1.1.5 SR 3.3.1.1.12	NA
	5[(a)]	1	H	SR 3.3.1.1.5 SR 3.3.1.1.12	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater Pump and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Four channels of Feedwater Pump and Main Turbine High Water Level Trip Instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more feedwater pump and main turbine high water level trip channels inoperable.	A.1 Place channel in trip.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. Feedwater pump and main turbine high water level trip capability not maintained.	B.1 Restore feedwater pump and main turbine high water level trip capability.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1	----------NOTE------------------- Only applicable if inoperable channel is the result of inoperable feedwater pump breaker or main turbine stop valve. -------------------------------------	
		Remove affected feedwater pump(s) and main turbine valve(s) from service.	4 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

---NOTE---

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 for up to 6 hours provided feedwater pump and main turbine high water level trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.2.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.2.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.2.3	Calibrate the trip units.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.2.4	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 49 inches.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.2.5	Perform LOGIC SYSTEM FUNCTIONAL TEST including valve and breaker actuation.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

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

LCO 3.3.4.1 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level - Low Low; 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. <u>OR</u>	14 days <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	---------------NOTE-------------- Not applicable if inoperable channel is the result of an inoperable breaker. ------------------------------------- Place channel in trip.	 14 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. One Function with ATWS-RPT trip capability not maintained.	B.1	Restore ATWS-RPT trip capability.	72 hours
C. Both Functions with ATWS-RPT trip capability not maintained.	C.1	Restore ATWS-RPT trip capability for one Function.	1 hour
D. Required Action and associated Completion Time not met.	D.1	---------------NOTE-------------- Only applicable if inoperable channel is the result of an inoperable breaker. ------------------------------------- Remove the affected recirculation pump from service.	 6 hours
	<u>OR</u>		
	D.2	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

--NOTE--

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 for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	--------------------------------NOTE-------------------------------- Not required for the time delay portion of the Reactor Vessel Water Level - Low Low Function. --- Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.1.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.1.3	Calibrate the trip units.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.1.4	Perform CHANNEL CALIBRATION of Reactor Vessel Water Level - Low Low time delay relays. The Allowable Value shall be ≥ 6 seconds and ≤ 8.6 seconds.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.5	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level – Low Low ≥ -48 inches; and b. Reactor Vessel Steam Dome Pressure - High ≤ 1155 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.1.6	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.2	---------------NOTE-------------- Only applicable for Functions 3.a and 3.b. -------------------------------------	
		Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	<u>AND</u>		
	B.3	Place channel in trip.	24 hours <u>OR</u> -----------NOTE---------- Not applicable when a loss of function occurs. ----------------------------- In accordance with the Risk Informed Completion Time Program
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.c, 1.d, 1.e, 1.f, 2.c, 2.d, 2.e, 2.i, 2.j, 2.l, and 2.m. -------------------------------------	
		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>		

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	C.2	Restore channel to OPERABLE status.	24 hours OR -----------NOTE---------- Not applicable when a loss of function occurs. ----------------------------- 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 HPCI pump suction is not aligned to the suppression pool. ------------------------------------- Declare HPCI System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	AND		
	D.2.1	Place channel in trip.	24 hours OR -----------NOTE---------- Not applicable when a loss of function occurs. ----------------------------- In accordance with the Risk Informed Completion Time Program
	OR		

ACTIONS (continued)

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

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
	AND	
	F.2 Place channel in trip.	-----------NOTE--------- Risk Informed Completion Time Program not applicable to loss of function. -----------------------------
		96 hours or in accordance with the Risk Informed Completion Time Program from discovery of inoperable channel concurrent with HPCI or reactor core isolation cooling (RCIC) inoperable
		AND
		8 days or 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 Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	G.2 Restore channel to OPERABLE status.	-----------NOTE--------- Risk Informed Completion Time Program not applicable to loss of function. ----------------------------- 96 hours or in accordance with the Risk Informed Completion Time Program from discovery of inoperable channel concurrent with HPCI or RCIC inoperable <u>AND</u> 8 days or in accordance with the Risk Informed Completion Time Program
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

SURVEILLANCE REQUIREMENTS

---NOTES---

1. Refer to Table 3.3.5.1-1 to determine which SRs apply for each ECCS Function.

2. 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 3.c and 3.f; and (b) for up to 6 hours for Functions other than 3.c and 3.f provided the associated Function or the redundant Function maintains ECCS initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.5.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.1.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.1.3	Calibrate the trip unit.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.1.4	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.1.5	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.1.6	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.5.1.7	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.1.8	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.5.1.9	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level - Low Low	1, 2, 3	4[(a)]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ -48 inches
b. Drywell Pressure - High	1, 2, 3	4[(a)]	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.8	≤ 2 psig
c. Reactor Steam Dome Pressure - Low (Injection Permissive)	1, 2, 3	2	C	SR 3.3.5.1.2 SR 3.3.5.1.4[(b)(c)] SR 3.3.5.1.8	≥ 397 psig and ≤ 440 psig
d. Reactor Steam Dome Pressure Permissive - Low (Pump Permissive)	1, 2, 3	2	C	SR 3.3.5.1.2 SR 3.3.5.1.4[(b)(c)] SR 3.3.5.1.8	≥ 397 psig
e. Reactor Steam Dome Pressure Permissive - Bypass Timer (Pump Permissive)	1, 2, 3	2	C	SR 3.3.5.1.7 SR 3.3.5.1.8	≤ 18 minutes

(a) Also required to initiate the associated emergency diesel generator (EDG).

(b) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the Technical Requirements Manual (TRM).

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
f. Core Spray Pump Start - Time Delay Relay	1, 2, 3	1 per pump	C	SR 3.3.5.1.7 SR 3.3.5.1.8	≤ 15.86 seconds
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level - Low Low	1, 2, 3	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ -48 inches
b. Drywell Pressure - High	1, 2, 3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.8	≤ 2 psig
c. Reactor Steam Dome Pressure - Low (Injection Permissive)	1, 2, 3	2	C	SR 3.3.5.1.2 SR 3.3.5.1.4(b)(c) SR 3.3.5.1.8	≥ 397 psig and ≤ 440 psig
d. Reactor Steam Dome Pressure Permissive - Low (Pump Permissive)	1, 2, 3	2	C	SR 3.3.5.1.2 SR 3.3.5.1.4(b)(c) SR 3.3.5.1.8	≥ 397 psig

(b) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the TRM.

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System					
e. Reactor Steam Dome Pressure Permissive - Bypass Timer (Pump Permissive)	1, 2, 3	2	C	SR 3.3.5.1.7 SR 3.3.5.1.8	≤ 18 minutes
f. Low Pressure Coolant Injection Pump Start - Time Delay Relay	1, 2, 3	4 per pump	B	SR 3.3.5.1.7 SR 3.3.5.1.8	
Pumps A, B					≤ 5.33 seconds
Pumps C, D					≤ 10.59 seconds
g. Low Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)	1, 2, 3	1 per pump	E	SR 3.3.5.1.2 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 360 gpm and ≤ 745 gpm
h. Reactor Steam Dome Pressure - Low (Break Detection)	1, 2, 3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 873.6 psig and ≤ 923.4 psig
i. Recirculation Pump Differential Pressure - High (Break Detection)	1, 2, 3	4 per pump	C	SR 3.3.5.1.2 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 63.5 inches wc
j. Recirculation Riser Differential Pressure - High (Break Detection)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.7[(b)(c)] SR 3.3.5.1.8	≤ 100.0 inches wc

(b) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the TRM.

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System					
k. Reactor Steam Dome Pressure - Time Delay Relay (Break Detection)	1, 2, 3	2	B	SR 3.3.5.1.7 SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 2.97 seconds
l. Recirculation Pump Differential Pressure - Time Delay Relay (Break Detection)	1, 2, 3	2	C	SR 3.3.5.1.7 SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 0.75 seconds
m. Recirculation Riser Differential Pressure - Time Delay Relay (Break Detection)	1, 2, 3	2	C	SR 3.3.5.1.7 SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 0.75 seconds
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level - Low Low	1, 2(d), 3(d)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ -48 inches
b. Drywell Pressure - High	1, 2(d), 3(d)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.8	≤ 2 psig
c. Reactor Vessel Water Level - High	1, 2(d), 3(d)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.7 SR 3.3.5.1.8	≤ 48 inches
d. Condensate Storage Tank Level - Low	1, 2(d), 3(d)	2	D	SR 3.3.5.1.7 SR 3.3.5.1.8	≥ 29.3 inches
e. Suppression Pool Water Level - High	1, 2(d), 3(d)	2	D	SR 3.3.5.1.5 SR 3.3.5.1.6 SR 3.3.5.1.8	≤ 3.0 inches
f. High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.5 SR 3.3.5.1.6 SR 3.3.5.1.8	≥ 362 gpm and ≤ 849 gpm

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level - Low Low	1, 2[(d)], 3[(d)]	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ -48 inches
b. Automatic Depressurization System Initiation Timer	1, 2[(d)], 3[(d)]	1	G	SR 3.3.5.1.7 SR 3.3.5.1.8	≤ 120 seconds
c. Core Spray Pump Discharge Pressure - High	1, 2[(d)], 3[(d)]	2	G	SR 3.3.5.1.2 SR 3.3.5.1.4[(b)(c)] SR 3.3.5.1.8	≥ 75 psig and ≤ 125 psig
d. Low Pressure Coolant Injection Pump Discharge Pressure - High	1, 2[(d)], 3[(d)]	4	G	SR 3.3.5.1.2 SR 3.3.5.1.4[(b)(c)] SR 3.3.5.1.8	≥ 75 psig and ≤ 125 psig
5. ADS Trip System B					
a. Reactor Vessel Water Level - Low Low	1, 2[(d)], 3[(d)]	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.7 SR 3.3.5.1.8	≥ -48 inches
b. Automatic Depressurization System Initiation Timer	1, 2[(d)], 3[(d)]	1	G	SR 3.3.5.1.7 SR 3.3.5.1.8	≤ 120 seconds

(b) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the TRM.

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 6 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. ADS Trip System B					
c. Core Spray Pump Discharge Pressure - High	1, 2[(d)], 3[(d)]	2	G	SR 3.3.5.1.2 SR 3.3.5.1.4[(b)(c)] SR 3.3.5.1.8	≥ 75 psig and ≤ 125 psig
d. Low Pressure Coolant Injection Pump Discharge Pressure - High	1, 2[(d)], 3[(d)]	4	G	SR 3.3.5.1.2 SR 3.3.5.1.4[(b)(c)] SR 3.3.5.1.8	≥ 75 psig and ≤ 125 psig

(b) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the nominal trip setpoint; otherwise, the channel shall be declared inoperable. The nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified in the TRM.

(d) With reactor steam dome pressure > 150 psig.

3.3 INSTRUMENTATION

3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation

LCO 3.3.5.2 The RCIC System instrumentation for each Function in Table 3.3.5.2-1 shall be OPERABLE.

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

ACTIONS

--NOTE--

Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.2-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	B.1 Declare RCIC System inoperable.	1 hour from discovery of loss of RCIC initiation capability
	<u>AND</u>	
	B.2 Place channel in trip.	24 hours <u>OR</u> -----------NOTE---------- Not applicable when a loss of function occurs. ----------------------------- In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	C.1	Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.2-1.	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
	AND		
	D.2.1	Place channel in trip.	24 hours OR -----------NOTE--------- Not applicable when a loss of function occurs. ---------------------------- In accordance with the Risk Informed Completion Time Program
	OR		
	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

---NOTES---

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. 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.	12 hours or in accordance with the Risk Informed Completion Time Program for Functions 2.a, 2.b, 5.c, 6.b, 7.a, and 7.b <u>AND</u> 24 hours or in accordance with the Risk Informed Completion Time Program for Functions other than Functions 2.a, 2.b, 5.c, 6.b, 7.a, and 7.b
B. One or more Functions with primary containment isolation capability not maintained.	B.1 Restore primary containment isolation capability.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1 Isolate associated main steam line (MSL).	12 hours
	<u>OR</u>	
	D.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	D.2.2 Be in MODE 4.	36 hours
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1 Be in MODE 2.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1 Isolate the affected penetration flow path(s).	1 hour
G. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	G.1 Isolate the affected penetration flow path(s).	24 hours
H. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	H.1 Declare associated standby liquid control (SLC) subsystem inoperable.	1 hour
	<u>OR</u>	
	H.2 Isolate the Reactor Water Cleanup System.	1 hour

3.3 INSTRUMENTATION

3.3.7.2 Mechanical Vacuum Pump Isolation Instrumentation

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

APPLICABILITY: MODES 1 and 2 with the 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.	12 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
	<u>OR</u>	
	A.2 --------------NOTE--------------- Not applicable if inoperable channel is the result of an inoperable mechanical vacuum pump breaker or isolation valve. ------------------------------------- Place channel in trip.	12 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Mechanical vacuum pump isolation capability not maintained.	B.1 Restore mechanical vacuum pump isolation capability	1 hour
C. Required Action and associated Completion Time not met.	C.1 Isolate the mechanical vacuum pump.	12 hours
	<u>OR</u>	
	C.2 Isolate the main steam lines.	12 hours
	<u>OR</u>	
	C.3 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

---NOTE---

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 for up to 6 hours provided the associated Function maintains mechanical vacuum pump isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.2.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.3	Perform CHANNEL CALIBRATION. The Allowable Value Shall be ≤ 6.9 R/hour.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST, including mechanical vacuum pump breaker and isolation valves actuation.	In accordance with the Surveillance Frequency Control Program

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

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.	1 hour <u>OR</u> -----------NOTE---------- Not applicable when a loss of function occurs. ----------------------------- In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Declare associated EDG inoperable.	Immediately

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (S/RVs)

LCO 3.4.3 The safety function of seven S/RVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required S/RVs inoperable.	A.1 Restore the required S/RVs to OPERABLE status.	14 days OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time of Condition A not met. OR Three or more required S/RVs inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

3.5 EMERGENCY CORE COOLING SYSTEM (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 three safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and 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 HPCI.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One LPCI pump inoperable.	A.1 Restore LPCI pump to OPERABLE status.	30 days
B. One LPCI subsystem inoperable for reasons other than Condition A. <u>OR</u> One Core Spray subsystem inoperable.	B.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One LPCI pump in both LPCI subsystems inoperable.	C.1	Restore one LPCI pump to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
D. Two LPCI subsystems inoperable for reasons other than Condition C or G.	D.1	Restore one LPCI subsystem to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
E. One Core Spray subsystem inoperable. AND One LPCI subsystem inoperable. OR One or two LPCI pump(s) inoperable.	E.1	Restore Core Spray subsystem to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
	OR		
	E.2	Restore LPCI subsystem to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
	OR		

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	E.3 Restore LPCI pump(s) to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	12 hours 36 hours
G. Two LPCI subsystems inoperable due to open RHR intertie return line isolation valve(s).	G.1 Isolate the RHR intertie line.	18 hours
H. Required Action and associated Completion Time of Condition G not met.	H.1 Be in MODE 2.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. HPCI System inoperable.	I.1 Verify by administrative means RCIC System is OPERABLE. <u>AND</u> I.2 Restore HPCI System to OPERABLE status.	Immediately 14 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
J. HPCI System inoperable. <u>AND</u> Condition A, B, or C entered.	J.1 Restore HPCI System to OPERABLE status. <u>OR</u> J.2 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program 72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
K. One ADS valve inoperable.	K.1 Restore ADS valve to OPERABLE status.	14 days <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. Required Action and associated Completion Time of Condition I, J, or K not met. <u>OR</u> One ADS valve inoperable and Condition A, B, C, D, or G entered. <u>OR</u> Two or more ADS valves inoperable. <u>OR</u> HPCI System inoperable and Condition D, E, or G entered.	L.1 Be in MODE 3. <u>AND</u> L.2 Reduce reactor steam dome pressure to $\leq$ 150 psig.	12 hours 36 hours
M. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition C, D, E, or G. <u>OR</u> HPCI System and one or more ADS valves inoperable.	M.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	-------------------------------NOTE------------------------------ Not required to be met for system vent flow paths opened under administrative control. -- Verify each ECCS injection/spray subsystem manual, power operated, and automatic valves in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify ADS pneumatic pressure is as follows for each required ADS pneumatic supply: a. S/RV Accumulator Bank header pressure ≥ 88.3 psig; and b. Alternate Nitrogen System pressure is ≥ 1060 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	-------------------------------NOTE------------------------------ Only required to be met in MODE 1. -- Verify the RHR System intertie return line isolation valves are closed.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.5	Verify correct breaker alignment to the LPCI swing bus.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	Verify each recirculation pump discharge valve cycles through one complete cycle of full travel or is de-energized in the closed position.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.1.7	Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor to containment pressure. System / Flow Rate / No. of Pumps / System Head Corresponding to a Reactor to Containment Pressure of Core Spray / ≥ 2835 gpm / 1 / ≥ 130 psi LPCI / ≥ 3870 gpm / 1 / ≥ 20 psi	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.1.8	-------------------------------NOTE------------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor steam dome pressure ≤ 1025.3 psig and ≥ 950 psig, the HPCI pump can develop a flow rate ≥ 2700 gpm against a system head corresponding to reactor pressure.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.9	-------------------------------NOTE------------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 2700 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.10	-------------------------------NOTE------------------------------ Vessel injection/spray may be excluded. --- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.11	-------------------------------NOTE------------------------------ Valve actuation may be excluded. --- Verify the ADS actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.12	-------------------------------NOTE------------------------------ Not required to be performed until 12 hours after reactor steam flow is adequate to perform the test. --- Verify each ADS valve is capable of being opened.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.13	Verify automatic transfer capability of the LPCI swing bus power supply from the normal source to the backup source.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEM (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 the RCIC System.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCIC System inoperable.	A.1 Verify by administrative means High Pressure Coolant Injection System is OPERABLE. <u>AND</u> A.2 Restore RCIC System to OPERABLE status.	Immediately 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. <u>AND</u> B.2 Reduce reactor steam dome pressure to $\leq$ 150 psig.	12 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.3	---------------NOTE-------------- Air lock doors in high radiation areas or areas with limited access due to inerting may be verified locked closed by administrative means. ------------------------------------- Verify an OPERABLE door is locked closed.	 Once per 31 days
C. Primary containment air lock 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
	<u>AND</u>		
	C.2	Verify a door is closed.	1 hour
	<u>AND</u>		
	C.3	Restore air lock to OPERABLE status.	24 hours <u>OR</u> 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
	<u>AND</u>		
	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, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

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 PCIVs. -------------------------------- One or more penetration flow paths with one PCIV inoperable for reasons other than Condition D or E.	A.1 AND	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours 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

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	--------------NOTES------------ 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------ Verify the affected penetration flow path is isolated.	 Once per 31 days following isolation 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

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ------------NOTE------------ Only applicable to penetration flow paths with two PCIVs. -------------------------------- One or more penetration flow paths with two PCIVs inoperable for reasons other than Condition D or E.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. ------------NOTE------------ Only applicable to penetration flow paths with only one PCIV. -------------------------------- One or more penetration flow paths with one PCIV inoperable.	C.1 <u>AND</u>	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	4 hours except for excess flow check valves (EFCVs) and penetrations with a closed system <u>AND</u> 72 hours for EFCVs and penetrations with a closed system

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	C.2 --------------NOTES------------ 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------ Verify the affected penetration flow path is isolated.	Once per 31 days following isolation 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

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more penetration flow paths with one or more 18 inch primary containment purge and vent valves not within purge and vent valve leakage limits.	D.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 when a loss of function occurs. ----------------------------- In accordance with the Risk Informed Completion Time Program
	<u>AND</u>		
	D.2	--------------NOTES------------ 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------ Verify the affected penetration flow path is isolated.	Once per 31 days following isolation for isolation devices outside containment

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One or more MSIVs with leakage rate not within limits.	E.1 Restore leakage rate to within limits.	8 hours
F. Required Action and associated Completion Time of Condition A, B, C, or D not met.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.1	-------------------------------NOTE------------------------------ Not required to be met when the 18 inch primary containment purge and vent valves are open for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open. --- Verify each 18 inch primary containment purge and vent valve is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.2	------------------------------NOTES----------------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. --- Verify each primary containment isolation manual valve and blind flange that is located outside primary containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.3	-------------------------------NOTES------------------------------ 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. --- Verify each primary containment manual isolation valve and blind flange that is located inside primary containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	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
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.5	Verify the isolation time of each power operated automatic PCIV, except for MSIVs, is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.6	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 9.9 seconds.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV actuates on a simulated instrument line break to restrict flow to ≤ 2 gpm.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.9	Verify each 18 inch primary containment purge and vent valve is blocked to restrict the valve from opening > 40°.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.10	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.11	Perform leakage rate testing for each 18 inch primary containment purge and vent valve with resilient seals.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.12	Verify leakage rate through each MSIV is: (a) ≤ 100 scfh when tested at ≥ 44.1 psig (P_a); or (b) ≤ 75.3 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.13	Verify leakage rate through the main steam pathway is: (a) ≤ 200 scfh when tested at ≥ 44.1 psig (P_a); or (b) ≤ 150.6 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.1.6 Reactor Building-to-Suppression Chamber Vacuum Breakers

LCO 3.6.1.6 Each reactor building-to-suppression chamber vacuum breaker shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more lines with one reactor building-to-suppression chamber vacuum breaker not closed.	A.1 Close the open vacuum breaker.	72 hours
B. One or more lines with two reactor building-to-suppression chamber vacuum breakers not closed.	B.1 Close one open vacuum breaker.	1 hour
C. One line with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	C.1 Restore the vacuum breaker(s) to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two lines with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	D.1 Restore all vacuum breakers in one line to OPERABLE status.	1 Hour
E. Required Action and Associated Completion Time not met.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.6.1	-------------------------------NOTES----------------------------- 1. Not required to be met for vacuum breakers that are open during Surveillances. 2. Not required to be met for vacuum breakers open when performing their intended function. --- Verify each vacuum breaker is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.6.2	Perform a functional test of each vacuum breaker.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.6.3	Verify the opening setpoint of each vacuum breaker is $\leq$ 0.5 psid.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.1.7 Suppression Chamber-to-Drywell Vacuum Breakers

LCO 3.6.1.7 Seven suppression chamber-to-drywell vacuum breakers shall be OPERABLE for opening.

AND

Eight suppression chamber-to-drywell vacuum breakers shall be closed.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required suppression chamber-to-drywell vacuum breaker inoperable for opening.	A.1 Restore one vacuum breaker to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
B. One suppression chamber-to-drywell vacuum breaker not closed.	B.1 Close the open vacuum breaker.	12 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

3.6.1.8 Residual Heat Removal (RHR) Drywell Spray

LCO 3.6.1.8 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 <u>OR</u> 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 <u>AND</u>	Be in MODE 3.	12 hours
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.8.1	Verify each RHR drywell spray subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.8.2	Verify each drywell spray header and nozzle is unobstructed.	10 years
SR 3.6.1.8.3	Verify RHR drywell spray subsystem locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pooling 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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.3.1	Verify each RHR suppression pool cooling subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.2.3.2	Verify each required RHR pump develops a flow rate ≥ 3870 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.2.3.3	Verify RHR suppression pool cooling subsystem locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.1 Residual Heat Removal Service Water (RHRSW) System

LCO 3.7.1 Two RHRSW subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHRSW subsystem inoperable.	A.1	---------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," for RHR shutdown cooling made inoperable by RHRSW System. ------------------------------------- Restore RHRSW subsystem to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Both RHRSW subsystems inoperable.	B.1	---------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.4.7 for RHR shutdown cooling made inoperable by RHRSW System. ------------------------------------- Restore one RHRSW subsystem to OPERABLE status.	 8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify each RHRSW manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.2 Emergency Service Water (ESW) System and Ultimate Heat Sink (UHS)

LCO 3.7.2 Two ESW subsystems and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ESW subsystem inoperable.	A.1	--------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," for RHR shutdown cooling made inoperable by ESW. ------------------------------------- Restore the ESW subsystem to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Both ESW subsystems inoperable. <u>OR</u> UHS inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	Verify the water level in the intake structure is ≥ 899 ft mean sea level.	In accordance with the Surveillance Frequency Control Program
SR 3.7.2.2	Verify the average water temperature of UHS is ≤ 90°F.	In accordance with the Surveillance Frequency Control Program
SR 3.7.2.3	-------------------------------NOTE------------------------------ Isolation of flow to individual components does not render ESW System inoperable. --- Verify each ESW subsystem manual and automatic valve in the flow paths servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.2.4	Verify each ESW subsystem actuates on an actual or simulated initiation signal.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.3	Restore required offsite circuit to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
B. One EDG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s), supported by the inoperable EDG, inoperable when 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 EDG is not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.2 for OPERABLE EDG.	24 hours
	AND		

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.4	Restore EDG to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required offsite circuit inoperable. AND One EDG inoperable.	--------------------NOTE------------------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "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 EDG to OPERABLE status.	12 hours OR In accordance with the Risk Informed Completion Time Program
E. Two EDGs inoperable.	E.1 Restore one EDG to OPERABLE status.	2 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3.	12 hours
	AND	
	F.2 Be in MODE 4.	36 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.2	------------------------------NOTES------------------------------ 1. All EDG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 2. A modified EDG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. --- Verify each EDG starts from standby conditions and achieves steady state voltage ≥ 3975 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	------------------------------NOTES------------------------------ 1. EDG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one EDG at a time. 4. This SR shall be preceded by and immediately follow, without shutdown, a successful performance of SR 3.8.1.2. -- Verify each EDG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2250 kW and ≤ 2500 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.4	Check for and remove accumulated water from each day tank and base tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.5	Verify the fuel oil transfer system operates to transfer fuel oil from the storage tank to the day tanks and from each day tank to the associated base tank.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.6	-------------------------------NOTE------------------------------ This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. -- Verify automatic and manual transfer of unit power supply from the normal offsite circuit to the alternate offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.7	------------------------------NOTES----------------------------- 1. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 2. If performed with EDG synchronized with offsite power, it shall be performed within the power factor limit. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. --- Verify each EDG rejects a load greater than or equal to its associated single largest post-accident load, and following load rejection, the frequency is ≤ 67.5 Hz.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.8	-------------------------------NOTE------------------------------ This Surveillance shall not normally be performed in MODE 1 or 2. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---	
	Verify on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal, permanently connected loads remain energized from the offsite power system and emergency loads are auto-connected through the time delay relays from the offsite power system.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.9	------------------------------NOTES----------------------------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 3. If part b is performed with EDG synchronized with offsite power, it shall be performed within the power factor limit. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ---	
	Verify each EDG operates for ≥ 8 hours: a. For ≥ 2 hours loaded ≥ 2625 kW and ≤ 2750 kW; and b. For the remaining hours of the test loaded ≥ 2250 kW and ≤ 2500 kW.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	------------------------------NOTES------------------------------ 1. This Surveillance shall be performed within 5 minutes of shutting down the EDG after the EDG has operated ≥ 2 hours loaded ≥ 2250 kW and ≤ 2500 kW. Momentary transients outside of load range do not invalidate this test. 2. All EDG starts may be preceded by an engine prelube period. ---	
	Verify each EDG starts and achieves: a. In ≤ 10 seconds, voltage ≥ 3975 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 3975 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.11	------------------------------NOTE------------------------------ This Surveillance shall not normally be performed in MODE 1, 2, or 3. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---	
	Verify each EDG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-------------------------------NOTES------------------------------ 1. All EDG starts may be preceded by an engine prelube period. 2. This Surveillance shall not normally be performed in MODE 1, 2, or 3. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --	
	Verify, on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. EDG auto-starts from standby condition and: 1. Energizes permanently connected loads in ≤ 10 seconds; 2. Energizes auto-connected emergency loads through time delay relays; 3. Achieves steady state voltage ≥ 3975 V and ≤ 4400 V; 4. Achieves steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz; and 5. Supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	-------------------------------NOTE------------------------------ This Surveillance shall not normally be performed in MODE 1, 2, or 3. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --- Verify interval between each sequenced load block is greater than or equal to the minimum design load interval.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 The Division 1 and Division 2 125 VDC and 250 VDC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required battery chargers on Division 1 or Division 2 inoperable.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	2 hours
	AND	
	A.2 Verify battery float current $\leq$ 2 amps for 250 VDC batteries and $\leq$ 1 amp for 125 VDC batteries.	Once per 12 hours
	AND	
	A.3 Restore required Division 1 or Division 2 battery charger(s) to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. One Division 1 or Division 2 DC electrical power subsystem inoperable for reasons other than Condition A.	B.1 Restore Division 1 or Division 2 DC electrical power subsystem to OPERABLE status.	2 hours OR In accordance with the Risk Informed Completion Time Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Distribution Systems - Operating

LCO 3.8.7 Division 1 and Division 2 AC and DC electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystems inoperable.	--------------------NOTE------------------- Enter applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," for DC divisions made inoperable by inoperable power distribution subsystems. ---	
	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. One or more DC electrical power distribution subsystems inoperable.	B.1 Restore DC electrical power distribution subsystem(s) to OPERABLE status.	2 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours
D. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct breaker alignments and voltage to required AC and DC electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

5.5 Programs and Manuals

5.5.14 Spent Fuel Pool Boral Monitoring Program

The program provides routine monitoring and actions to ensure that the condition of Boral in the spent fuel pool racks is appropriately monitored to ensure that the Boral neutron attenuation capability described in the criticality safety analysis of USAR Section 10.2.1 is maintained. The program shall include the following:

a. Periodic physical examination of representative Boral coupons or in situ storage racks at a frequency defined by observed trends or calculated projections of Boral degradation. The measurement will be performed to ensure that average thickness of the coupon (or average thickness of a representative area of the in situ storage rack) does not exceed the nominal design thickness of the coupon (or storage rack) plus the 0.055-inch dimension assumed for the analyzed blister.

b. Neutron attenuation testing of a representative Boral coupon or in situ storage rack shall be performed prior to December 31, 2015, and thereafter at a frequency of not more than 10 years, or more frequently based on observed trends or calculated projections of Boral degradation. The acceptance criterion for minimum boron areal density will be that value assumed in the criticality safety analysis (0.013 gm/cm^2).

c. Description of appropriate corrective actions for discovery of nonconforming Boral.

5.5.15 Surveillance Frequency Control Program

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

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

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

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

5.5 Programs and Manuals

5.5.16 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. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 206 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By application dated March 30, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20090F820), as supplemented by letters dated December 21, 2020 (ADAMS Accession No. ML20356A131), April 20, 2021 (ADAMS Accession No. ML21110A666), and June 30, 2021 (ADAMS Accession No. ML21181A308) Northern States Power Company, doing business as Xcel Energy (NSPM, the licensee) submitted a license amendment request (LAR) for Monticello Nuclear Generating Plant (MNGP).

The amendment would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation (LCOs) are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated July 2, 2018 (ADAMS Package Accession No. ML18183A493). The U.S. Nuclear Regulatory Commission (NRC or the Commission) issued a final model safety evaluation (SE) approving TSTF-505, Revision 2, on November 21, 2018 (ADAMS Package Accession No. ML18269A041).

The LAR proposed variations from the TS changes described in TSTF-505, Revision 2. The variations are described in Section 2.2.4 of this SE.

The supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on May 19, 2020 (85 FR 29985).

2.0 REGULATORY EVALUATION

2.1 Description of Risk Informed Completion Time (RICT) Program

The TS LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, the licensee must shut down the reactor or follow any remedial or required action (e.g., testing, maintenance, or repair activity) permitted by the TSs until the condition can be met. The remedial actions (i.e., ACTIONS) associated with an LCO contain Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times (CTs). The CTs are referred to as the "front stops" in the context of this SE. For certain Conditions, the TS require exiting the Mode of Applicability of an LCO (i.e., shutdown the reactor).

The Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09-A, Revision 0, "Risk Informed Technical Specifications Initiative 4b: Risk Managed Technical Specifications (RMTS)," dated November 2006 (NEI 06-09 A) (ADAMS Accession No. ML122860402) provides a methodology for extending existing CTs and thereby delaying exiting the operational mode of applicability or taking Required Actions if risk is assessed and managed within the limits and programmatic requirements established by a RICT program.

2.2 Description of TS Changes

The amendment requested approval to add a RICT program to the Administrative Controls section of the TS and to modify selected CTs to permit extending the CTs, provided risk is assessed and managed as described in NEI 06-09-A. The LAR proposed to use NEI 06-09-A and included documentation regarding the technical adequacy of the probabilistic risk assessment (PRA) models for the RICT program, consistent with the guidance of Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk informed Activities," March 2009 (ADAMS Accession No. ML090410014).

2.2.1 TS 1.0, “Use and Application”

Example 1.3-8 would be added to TS 1.3, “Completion Times,” and reads as follows:

<u>EXAMPLE 1.3-8</u>

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.

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, Conditions A and B are exited, and therefore, the required actions of Condition B may be terminated.

2.2.2 TS 5.5.16 - Risk informed Completion Time Program

TS 5.5.16, which describes the RICT program, would be added to the TS and reads as follows:

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. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

2.2.3 Application of the RICT program to Existing LCOs and Conditions

The typical CT is modified by the application of the RICT program as shown in the following example. The changed portion is indicated in italics.

ACTIONS

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

Where necessary, conforming changes are made to CTs to make them accurate following use of a RICT. For example, most TSs have requirements to close/isolate containment isolation devices if one or more containment penetrations have inoperable devices. This is followed by a requirement to periodically verify the penetration is isolated. By adding the flexibility to use a

RICT to determine a time to isolate the penetration, the periodic verifications must then be based on the time “following isolation.”

Individual LCO Required Actions and CTs modified by the proposed change are identified below.

TS 3.1.7 – Standby Liquid Control (SLC) System

- Action B: With one SLC subsystem inoperable for reasons other than Condition A, restore SLC subsystem to OPERABLE status within 7 days *or in accordance with the Risk Informed Completion Time Program*.

TS 3.3.1.1 – Reactor Protection System (RPS) Instrumentation

- Action A.1: With one or more required channels inoperable, place channel in trip within 12 hours *or in accordance with the Risk Informed Completion Time Program.*

- Action A.2: With one or more required channels inoperable, place associated trip system in trip within 12 hours *or in accordance with the Risk Informed Completion Time Program*.

- Action B.1: With one or more functions with one or more required channels inoperable in both trip systems, place channel in one trip system in trip within 6 hours *or in accordance with the Risk Informed Completion Time Program.*

- Action B.2: With one or more functions with one or more required channels inoperable in both trip systems, place one trip system in trip within 6 hours *or in accordance with the Risk Informed Completion Time Program.*

TS 3.3.2.2 – Feedwater Pump and Main Turbine High Water Level Trip Instrumentation

- Action A.1: With one or more feedwater pump and main turbine high water level trip channels inoperable, place channel in trip within 7 days *or in accordance with the Risk Informed Completion Time Program.*

TS 3.3.4.1 – Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

- Action A.1: With one or more channels inoperable, restore channel to OPERABLE status within 14 days *or in accordance with the Risk Informed Completion Time Program.*

- Action A.2: With one or more channels inoperable, place channel in trip within 14 days *or in accordance with the Risk Informed Completion Time Program.*

<u>TS 3.3.5.1 – Emergency Core Cooling Systems (ECCS) Instrumentation</u>

- Action B.3: With one Function with ATWS-RPT trip capability not maintained, place channel in trip within 24 hours *or in accordance with the Risk Informed Completion Time Program.*
  Note: Not applicable when a loss of function occurs.

- Action C.2: As required by Required Action A.1 and referenced in Table 3.3.5.1-1, restore channel to OPERABLE status within 24 hours *or in accordance with the Risk Informed Completion Time Program.*
  Note: Not applicable when a loss of function occurs.

- Action D.2.1: As required by Required Action A.1 and referenced in Table 3.3.5.1-1, place channel in trip within 24 hours *or in accordance with the Risk Informed Completion Time Program.*
- Note: Not applicable when a loss of function occurs.

- Action E.2: As required by Required Action A.1 and referenced in Table 3.3.5.1-1, restore channel to OPERABLE status within 7 days *or in accordance with the Risk Informed Completion Time Program.*
- Note: Not applicable when a loss of function occurs.

- Action F.2: As required by Required Action A.1 and referenced in Table 3.3.5.1-1, place channel in trip within 96 hours from discovery of inoperable channel concurrent with HPCI [high pressure coolant injection] or reactor core isolation cooling (RCIC) inoperable *or in accordance with the Risk Informed Completion Time Program* AND within 8 days *or in accordance with the Risk Informed Completion Time Program.*
  Note: Risk Informed Completion Time Program not applicable to loss of function.

- Action G.2: As required by Required Action A.1 and referenced in Table 3.3.5.1-1, restore channel to OPERABLE status within 96 hours from discovery of inoperable channel concurrent with HPCI or (RCIC) inoperable *or in accordance with the Risk Informed Completion Time Program* AND within 8 days *or in accordance with the Risk Informed Completion Time Program.*
  Note: Risk Informed Completion Time Program not applicable to loss of function.

<u>TS 3.3.5.2 – Reactor Core Isolation Cooling (RCIC) System Instrumentation</u>

- Action B.2: As required by Required Action A.1 and referenced in Table 3.3.5.2-1, place channel in trip within 24 hours *or in accordance with the Risk Informed Completion Time Program.*
  Note: Not applicable when a loss of function occurs.

- Action D.2.1 As required by Required Action A.1 and referenced in Table 3.3.5.2-1, place channel in trip within 24 hours *or in accordance with the Risk Informed Completion Time Program.*
  Note: Not applicable when a loss of function occurs.

TS 3.3.6.1 – Primary Containment Isolation Instrumentation

- Action A.1: With one or more required channels inoperable, place channel in trip within 12 hours *or in accordance with the Risk Informed Completion Time Program* for Functions 2.a, 2.b, 5.c, 6.b, 7.a, and 7.b AND within 24 hours *or in accordance with the Risk Informed Completion Time Program* for Functions other than Functions 2.a, 2.b, 5.c, 6.b, 7.a, and 7.b.

TS 3.3.8.1 – Loss of Power (LOP) Instrumentation

- Action A.1: With one or more channels inoperable, place channel in trip within 1 hour *or in accordance with the Risk Informed Completion Time Program*. Note: Not applicable when a loss of function occurs.

TS 3.4.3 – Safety/Relief Valves (S/RVs)

- Action A.1: With one or two required S/RVs inoperable, restore the required S/RVs to OPERABLE status within 14 days *or in accordance with the Risk Informed Completion Time Program*.

TS 3.5.1 – ECCS – Operating

- Action I.2: With HPCI [high-pressure cooling injection] System inoperable, restore HPCI System to OPERABLE status within 14 days *or in accordance with the Risk Informed Completion Time Program*.

- Action J.1: With HPCI System inoperable AND Condition A, B, or C entered, restore HPCI System to OPERABLE status within 72 hours *or in accordance with the Risk Informed Completion Time Program*.

- Action J.2: With HPCI System inoperable AND Condition A, B, or C entered, restore low pressure ECCS injection/spray subsystem(s) toOPERABLE status within 72 hours *or in accordance with the Risk Informed Completion Time Program*.

- Action K.1: With one ADS [automatic depressurization system] valve inoperable, restore ADS valve to OPERABLE status within 14 days *or in accordance with the Risk Informed Completion Time Program*.

TS 3.5.3 – RCIC System

- Action A.2: With RCIC System inoperable, restore RCIC System to OPERABLE status within 14 days *or in accordance with the Risk Informed Completion Time Program*.

TS 3.6.1.2 – Primary Containment Air Lock

- Action C.3: With primary containment air lock inoperable for reasons other than Condition A or B, restore air lock to OPERABLE status within 24 hours *or in accordance with the Risk Informed Completion Time Program*.

TS 3.6.1.3 – Primary Containment Isolation Valves (PCIVs)

- Action A.1: With one or more penetration flow paths with one PCIV inoperable for reasons other than Condition D or E, 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 within 4 hours except for main steam line *or in accordance with the Risk Informed Completion Time Program* AND within 8 hours for main steam line *or in accordance with the Risk Informed Completion Time Program*.

- Action A.2: The CT for the Required Action to verify the affected penetration flow path is isolated, has been modified by adding the words “following isolation” after “once per 31 days.”

- Action C.2: The CT for the Required Action to verify the affected penetration flow path is isolated, has been modified by adding the words “following isolation” after “once per 31 days.”

- Action D.1: With one or more penetration flow paths with one or more 18 inch primary containment purge and vent valves not within purge and vent valve leakage limits, isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange within 24 hours *or in accordance with the Risk Informed Completion Time Program*.
  Note: Not applicable when a loss of function occurs.

- Action D.2: The CT for the Required Action to verify the affected penetration flow path is isolated, has been modified by adding the words “following isolation” after “once per 31 days.”

TS 3.6.1.6 – Reactor Building-to-Suppression Chamber Vacuum Breakers

- Action C.1: With one line with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening, restore the vacuum breaker(s) to OPERABLE status within 72 hours *or in accordance with the Risk Informed Completion Time Program*.

TS 3.6.1.7 – Suppression Chamber-to-Drywell Vacuum Breakers

- Action A.1: With one required suppression chamber-to-drywell vacuum breaker inoperable for opening, restore one vacuum breaker to OPERABLE status within 72 hours *or in accordance with the Risk Informed Completion Time Program*.

TS 3.6.2.3 – Residual Heat Removal (RHR) Suppression Pool Cooling

- Action A.1: With one RHR suppression pool cooling subsystem inoperable, restore RHR suppression pool cooling subsystem to OPERABLE status within 7 days *or in accordance with the Risk Informed Completion Time Program*.

TS 3.7.1 – Residual Heat Removal Service Water (RHRSW) System

- Action A.1: With one RHRSW subsystem inoperable, restore RHRSW subsystem to OPERABLE status within 7 days *or in accordance with the Risk Informed Completion Time Program*.

TS 3.7.2 – Emergency Service Water (ESW) System and Ultimate Heat Sink (UHS)

- Action A.1: With one ESW subsystem inoperable, restore the ESW subsystem to OPERABLE status within 72 hours *or in accordance with the Risk Informed Completion Time Program*.

TS 3.8.1 – AC Sources – Operating

- Action A.3: With one required offsite circuit inoperable, restore required offsite circuit to OPERABLE status within 72 hours *or in accordance with the Risk Informed Completion Time Program*.

- Action B.4: With one EDG [emergency diesel generator] inoperable, restore EDG to OPERABLE status within 7 days *or in accordance with the Risk Informed Completion Time Program*.

- Action C.2: With two required offsite circuits inoperable, restore one required offsite circuit to OPERABLE status within 24 hours *or in accordance with the Risk Informed Completion Time Program*.

- Action D.1: With one required offsite circuit inoperable AND one EDG inoperable, restore required offsite circuit to OPERABLE status within 12 hours *or in accordance with the Risk Informed Completion Time Program.*

- Action D.2: With one required offsite circuit inoperable AND one EDG inoperable, restore EDG to OPERABLE status within 12 hours *or in accordance with the Risk Informed Completion Time Program*.

TS 3.8.4 – DC Sources – Operating

- Action A.3: With one or more required battery chargers on Division 1 or Division 2 inoperable, restore required Division 1 or Division 2 battery charger(s) to OPERABLE status within 7 days *or in accordance with the Risk Informed Completion Time Program*.

- Action B.1: With one Division 1 or Division 2 DC electrical power subsystem inoperable for reasons other than Condition A, restore Division 1 or Division 2 DC electrical power subsystem to OPERABLE status within 2 hours *or in accordance with the Risk Informed Completion Time Program*.

TS 3.8.7 – Distribution Systems – Operating

- Action A.1: With one or more AC [alternating current] electrical power distribution subsystems inoperable, restore AC electrical power distribution subsystem(s) to OPERABLE status within 8 hours *or in accordance with the Risk Informed Completion Time Program*.

- Action B.1: With one or more DC [direct current] electrical power distribution subsystems inoperable, restore DC electrical power distribution subsystem(s) to OPERABLE status within 2 hours *or in accordance with the Risk Informed Completion Time Program*.

2.2.4 Variations from TSTF-505, Revision 2

2.2.4.1 Application of the RICT program to Modified Conditions, Required Actions, and Completion Times

The following Conditions are modified to permit the application of a RICT:

TS 3.5.1 – ECCS – Operating

- Action B.1: With one LPCI subsystem inoperable for reasons other than Condition A OR one core spray (CS) subsystem inoperable, restore low pressure ECCS injection/spray subsystem to OPERABLE status within 7 days *or in accordance with the Risk Informed Completion Time Program*.

- Action C.1: With one LPCI pump in both LPCI subsystems inoperable, restore one LPCI pump to OPERABLE status within 7 days *or in accordance with the Risk Informed Completion Time Program*.

- Action D.1: With two LPCI subsystems inoperable for reasons other than Condition C or G, restore one LPCI subsystem to OPERABLE status within 72 hours *or in accordance with the Risk Informed Completion Time Program*.

- Action E.1: With one CS subsystem inoperable AND one LPCI subsystem inoperable OR one or two LPCI pump(s) inoperable, restore CS subsystem to OPERABLE status within 72 hours *or in accordance with the Risk Informed Completion Time Program*.

- Action E.2: With one CS subsystem inoperable AND one LPCI subsystem inoperable OR one or two LPCI pump(s) inoperable, restore LPCI subsystem to OPERABLE status within 72 hours *or in accordance with the Risk Informed Completion Time Program*.

- Action E.3: With one CS subsystem inoperable AND one LPCI subsystem inoperable OR one or two LPCI pump(s) inoperable, restore LPCI pump(s) to

OPERABLE status within 72 hours *or in accordance with the Risk Informed Completion Time Program*.

TS 3.6.1.8 – Residual Heat Removal (RHR) Drywell Spray

- Action A.1: With one RHR drywell spray subsystem inoperable, restore RHR drywell spray subsystem to OPERABLE status within 7 days *or in accordance with the Risk Informed Completion Time Program*.

2.2.4.2 Application of the RICT to Additional ACTIONS Requirements

The following individual LCO Actions and CTs identified below are modified by the proposed change to permit the application of a RICT and are in addition to those included in TSTF-505.

TS 3.3.7.2 – Mechanical Vacuum Pump Isolation Instrumentation

- Action A.1: With one or more channels inoperable, restore channel to OPERABLE status within 12 hours *or in accordance with the Risk Informed Completion Time Program.*

- Action A.2: With one or more channels inoperable, place channel in trip within 12 hours *or in accordance with the Risk Informed Completion Time Program.*

2.2.4.3 Proposed Changes to TSs not associated with TSTF-505, Revision 2

Attachment 1, Section 2.4, "Optional Variations," items 8 and 9, of the LAR included a description of proposed administrative changes to the TSs. The proposed administrative changes include:

- alignment of text in TS 3.3.2.2, TS 3.3.7.2, TS 3.6.1.3, and TS 3.6.2.3
- correction of a typographical error in TS 3.3.5.1,
- deletion of reference to prior modification to TS 3.3.7.2, and
- correction of system title in TS 3.3.5.1

2.3 Regulatory Review

2.3.1 Applicable Regulations

Under 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," of Title 10 of the *Code of Federal Regulations* (10 CFR) whenever a holder of a license wishes to amend the license, including TSs in the license, an application for amendment must be filed, fully describing the changes desired. Under 10 CFR 50.92(a), determinations of whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses or construction permits to the extent applicable and appropriate.

The regulation under 10 CFR 50.36(c)(2) requires that TSs contain LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCO can be met. Typically, the TSs require restoration of equipment in a timeframe commensurate with its safety significance, along with other engineering considerations. The regulation under 10 CFR 50.36(b) requires that TSs be derived from the analyses and evaluation included in the safety analysis report and amendments thereto.

The regulation under 10 CFR 50.36(c)(5) states that administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

In determining whether the proposed TS remedial actions should be granted, the Commission will apply the "reasonable assurance" standards of 10 CFR 50.40(a) and 50.57(a)(3). The regulation at 10 CFR 50.40(a) states that in determining whether to grant the licensing request, the Commission will be guided by, among other things, consideration about whether "the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to any of the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in Part 20 of this chapter, and that the health and safety of the public will not be endangered."

The regulation under 10 CFR 50.55a(h) "Protection and safety systems" states that protection systems of nuclear power reactors of all types must meet the requirements specified in this paragraph. Each combined license for a utilization facility is subject to the conditions specified in this clause.

Section 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (i.e., the Maintenance Rule), requires licensees to monitor the performance or condition of SSCs against licensee-established goals in a manner sufficient to provide reasonable assurance that these SSCs are capable of fulfilling their intended functions. The regulation under 10 CFR 50.65(a)(4) requires the assessment and management of the increase in risk that may result from a proposed maintenance activity.

The regulation under 10 CFR 50.46, "Acceptance Criteria for emergency core cooling systems for light-water nuclear power reactor," requires that the ECCS be designed with sufficient margin to assure that the design safety limits specified in 10 CFR 50.46(b) are met during loss-of-coolant accidents (LOCAs).

The MNGP Updated Safety Analysis Report (USAR), Appendix E, "Plant Comparative Evaluation with the Proposed AEC [Atomic Energy Commission] 70 Design Criteria," (ADAMS Accession No. ML20003D166) states that the offsite power system and the onsite power system conform to the intent of the following draft General Design Criteria (GDC) for Nuclear Power Plant Construction Permits proposed by the AEC in July 1967:

- Criterion 24 - Emergency Power for Protection Systems (Category B) states: "In the event of the loss of all off-site power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems."

- Criterion 39 - Emergency Power for Engineered Safety Features (Category A) states: "Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system."

2.3.2 Regulatory Guidance

Revision 2 of RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-specific Changes to the Licensing Basis" May 2011 (ADAMS Accession No. ML100910006), describes an acceptable risk-informed approach for assessing the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights. This regulatory guide also provides risk acceptance guidelines for evaluating the results of such evaluations.

Revision 1 of RG 1.177, "An Approach for Plant-specific, Risk informed Decisionmaking: Technical Specifications," May 2011 (ADAMS Accession No. ML100910008), describes an acceptable risk-informed approach specifically for assessing proposed TS changes. This regulatory guide identifies a three-tiered approach for a licensee's evaluation of the risk associated with a proposed TS CT change, as follows.

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's [U.S. Nuclear Regulatory Commission (NRC or Commission] Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. The first tier assesses the impact on plant risk as expressed by the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out-of-service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). Tier 1 also addresses PRA acceptability, including the technical adequacy of the licensee's plant-specific PRA for the subject application.

- Tier 2 identifies and evaluates any potential risk significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is removed from service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that there are appropriate restrictions in place such that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed CT is implemented.

- Tier 3 addresses the licensee's Configuration Risk Management Program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk significant configurations that may not have been considered when the Tier 2 evaluation was performed. Compared with Tier 2, Tier 3 provides additional coverage to ensure risk significant plant equipment outage configurations are identified in a timely manner and that the risk impact of out of service equipment is appropriately evaluated prior to performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule, which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1 and the adequacy of the licensee's program and PRA model for this application. The CRMP ensures that equipment removed from service prior to or during the proposed extended CT will be appropriately assessed from a risk perspective.

Revision 2 of RG 1.200 describes an acceptable approach for determining whether the PRA acceptability, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors. This RG provides guidance for assessing the technical adequacy of a PRA. Revision 2 of RG 1.200, endorses, with clarifications and qualifications, the use of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard, RA-Sa-2009, "Addenda to ASME RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (i.e., the PRA Standard).

As discussed in RG 1.177, Revision 1, and RG 1.174, Revision 2, a risk informed application should be evaluated to ensure that the proposed changes meet the following key principles:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption;

2. The proposed change is consistent with the defense in depth (DID) philosophy;

3. The proposed change maintains sufficient safety margins;

4. When proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement; and

5. The impact of the proposed change should be monitored using performance measurement strategies.

3.0 TECHNICAL EVALUATION

The licensee's adoption of TSTF-505, Revision 2, provides for the addition of a RICT program to the Administrative Controls section of the TS and modifies selected Required Action CTs to permit extending the CTs, provided risk is assessed and managed as described in NEI 06-09-A. In accordance with NEI 06-09-A, PRA methods are used to justify each extension to a Required Action CT based on the specific plant configuration which exists at the time of the applicability of the Required Action and are updated when plant conditions change. The licensee's LAR included documentation regarding the technical adequacy of the PRA models used in the CRMP, consistent with the requirements of RG 1.200.

Most TS identify one or more Conditions for which the LCO may not be met, to permit a licensee to perform required testing, maintenance, or repair activities. Each Condition has an associated Required Action for restoration of the LCO or for other actions, each with some fixed time interval, referred to as the CT, which identifies the time interval permitted to complete the Required Action. Upon expiration of the CT, the licensee is required to shut down the reactor or follow the Required Action(s) stated in the ACTIONS requirements. The RICT program provides the necessary administrative controls to permit extension of CTs and thereby delay reactor shutdown or Required Actions, if risk is assessed and managed within specified limits and programmatic requirements. The specified safety function or performance level of TS required equipment is unchanged, and the Required Action(s), including the requirement to shut down the reactor, are also unchanged. Only the CTs for the Required Actions are extended by the RICT program.

The NRC staff reviewed the licensee's PRA methods and models to determine if they are technically acceptable for use in the proposed risk-informed completion time extensions. The NRC staff also reviewed the licensee's proposed RICT program to determine if it provides the necessary administrative controls to permit completion time extensions.

In August 2020, the NRC staff and its contractors from the Pacific Northwest National Laboratory participated in a virtual regulatory audit. The NRC staff performed the audit to ascertain the information needed to support its review of the application and develop requests for additional information (RAIs), as needed. On January 8, 2021, the NRC staff issued an audit summary (ADAMS Accession No. ML20328A038). By electronic mail dated October 26, 2020 (ADAMS Accession No. ML20302A197) and March 9, 2021 (ADAMS Accession No. ML21068A277), the NRC sent the licensee RAIs. By letters dated December 21, 2020, and April 20, 2021, the licensee responded to the RAIs.

3.1 Review of Key Principles

Revision 1 of RG 1.177 and RG 1.174, Revision 2, identify five key safety principles to be applied to risk informed changes to the TSs. Each of these principles are addressed in NEI 06-09-A. The NRC staff's evaluation of the licensee's proposed use of RICTs against these key safety principles is discussed below.

3.1.1 Key Principle 1: Evaluation of Compliance with Current Requirements

As stated in 10 CFR 50.36(c)(2):

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

When the necessary redundancy is not maintained (e.g., one train of a two-train system is inoperable), the TSs permit a limited period of time to restore the inoperable train to operable status and/or take other remedial measures. If these actions are not completed within the CT, the TSs normally require that the plant exit the mode of applicability for the LCO. With one train of a two-train system inoperable, the TS safety function is accomplished by the remaining operable train. In the current TSs, the CT is specified as a fixed time period (termed the "front stop"). The addition of the option to determine the CT in accordance with the RICT program would allow an evaluation to determine a configuration-specific CT. The evaluation would be done in accordance with the methodology prescribed in NEI 06-09-A and TS 5.5.16. The RICT is limited to a maximum of 30 days (termed the "back stop"). The CTs in the current TSs were established using experiential data, risk insights, and engineering judgment. The RICT program provides the necessary administrative controls to permit extension of CTs and thereby delay reactor shutdown or Required Actions, if risk is assessed and managed appropriately within specified limits and programmatic requirements.

When the necessary redundancy is not maintained, and the system loses the capability to perform its safety function(s) without any further failures (e.g., two trains of a two-train system are inoperable), the plant must exit the mode of applicability for the LCO, or take remedial actions, as specified in the TSs. A configuration-specific RICT may not be used in this condition. With the incorporation of the RICT program, the required performance levels of equipment specified in LCOs are not changed. Only the required CT for the Required Actions are modified by the RICT program.

3.1.1.1 Key Principle 1: Conclusions

Based on the discussion provided above, the NRC staff finds that the proposed changes meet the first key safety principle of RG 1.174, Revision 2, and RG 1.177, Revision 1.

3.1.2 Key Principle 2: Evaluation of Defense in Depth (DID)

DID is an approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. DID includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.

As discussed throughout RG 1.174, consistency with the DID philosophy is maintained by the following measures:

- Preserve a reasonable balance among the layers of defense.
- Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.
- Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.
- Preserve adequate defense against potential CCFs.
- Maintain multiple fission product barriers.
- Preserve sufficient defense against human errors.
- Continue to meet the intent of the plant's design criteria.

The proposed change represents a robust technical approach that preserves a reasonable balance among avoidance of core damage, avoidance of containment failure, and consequence mitigation. The three-tiered approach to risk informed TS CT changes provides additional assurance that DID will not be significantly impacted by such changes to the licensing basis. The licensee is proposing no changes to the design of the plant or any operating parameter, no new operating configurations, and no new changes to the design basis in the proposed changes to the TS.

The effect of the proposed changes when implemented will be that the RICT program will allow CTs to vary based on the risk significance of the given plant configuration (i.e., the equipment out of service at any given time) provided that the system(s) retain(s) the capability to perform the applicable safety function(s) without any further failures (e.g., one train of a two-train system is inoperable). A configuration-specific RICT may not be used if the system has lost the capability to perform its safety function(s). These restrictions on inoperability of all required trains of a system ensure that consistency with the DID philosophy is maintained by following existing guidance when the capability to perform TS safety function(s) is lost.

The proposed RICT program uses plant-specific operating experience for component reliability and availability data. Thus, the allowances permitted by the RICT program are directly reflective of actual component performance in conjunction with component risk significance. In some cases, the RICT program may use compensatory actions to reduce calculated risk in some configurations. Where credited in the PRA, these actions are incorporated into station procedures or work instructions and have been modeled using appropriate human reliability considerations. Application of the RICT program determines the risk significance of plant configurations. It also permits the operator to identify the equipment that has the greatest effect on the existing configuration risk. With this information, the operator can manage the out of service duration and determine the consequences of removing additional equipment from service.

The application of the RICT program places high value on key safety functions and works to ensure they remain a top priority over all plant conditions. The RICT will be applied to extend CTs on key electrical power distribution systems. Failures in electrical power distribution systems can simultaneously affect multiple safety functions; therefore, potential degradation to DID during the extended CTs is discussed further below.

3.1.2.1 Use of Compensatory Measures to Retain DID

Application of the RICT program provides a structure to assist the operator in identifying effective compensatory actions for various plant maintenance configurations to maintain and manage acceptable risk levels. NEI 06-09-A addresses potential compensatory actions and RMA measures by stating, in generic terms, that compensatory measures may include but are not limited to the following:

- Reduce the duration of risk-sensitive activities.
- Remove risk-sensitive activities from the planned work scope.
- Reschedule work activities to avoid high risk-sensitive equipment outages or maintenance states that result in high-risk plant configurations. Accelerate the restoration of out of service equipment.
- Determine and establish the safest plant configuration.

NEI 06-09-A requires that compensatory measures be initiated when the PRA calculated RMA time (RMAT) is exceeded, or for preplanned maintenance for which the RMAT is expected to be exceeded, RMAs shall be implemented at the earliest appropriate time.

3.1.2.2 Evaluation of Electrical Power Systems

According to the MNGP USAR Chapter 8, "Plant Electrical Systems," (ADAMS Accession No. ML20003D118), the plant is designed such that the safety functions are maintained assuming a single failure within the electrical power system. By incorporating an electrical power supply perspective, this concept is further reflected in a number of principal design criteria. Single-failure requirements are typically suspended for the time that a plant is not meeting an LCO (i.e., in an ACTION statement). This section considers the plant configurations from a DID perspective.

As stated in the MNGP USAR, Section 1.2.6 (ADAMS Accession No. ML20003D135), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the engineered safety feature (ESF) systems.

As described in the MNGP USAR Chapter 8, the MNGP offsite power is provided by three transformers (2R, 1R, and 1AR) that can independently provide adequate power for the plant's safety-related loads. The primary station auxiliary transformer 2R is fed from 345-kilovolt (kV) Bus No. 1 via 345 kV to 34.5 kV transformer 2RS. The reserve transformer 1R is fed from the 115 kV substation via an overhead line. The transformers 2R and 1R are each of adequate size to provide the plant's full auxiliary load requirements. The reserve auxiliary transformer, 1AR, may be fed from two separate 13.8 kV sources in the substation. One method of supplying the 1AR transformer is from the tertiary winding of the No. 10 transformer, the auto-transformer which interconnects the 34.5 kV and 115 kV systems. The alternate method of feeding 1AR is from the 345 kV substation via 345 kV/13.8 kV transformer 1ARS. The transformer 1AR is sized to provide only the plant's essential 4160 volts alternating current (VAC) and connected loads.

Two independent EDGs provide redundant standby power sources. The standby EDGs provide AC power to essential loads on the safety related 4160 VAC buses in the event of a loss or degradation of all off-site power sources. On loss of auxiliary power, the reactor will scram, and if auxiliary power is not restored immediately, the EDGs which have automatically started will carry the vital loads. Each EDG can provide enough power to safely shut down the reactor upon the loss of all outside power simultaneous with a DBA. The EDGs are each capable of starting and carrying the largest safe shutdown loads required under postulated accident conditions. The loading of each EDG unit is below its 2500-kilowatt (kW) continuous rating for the DBA LOCA including a loss of offsite AC power.

Two motor generator sets supply AC power to the RPS. These sets are powered from 480 VAC buses and are used to supply power to the RPS scram logic channels as well as neutron and radiation monitoring systems.

Two independent divisions, each including 250-VDC (volt direct current) and 125-VDC battery systems, are provided. The 250-VDC batteries are sized to provide adequate voltage at the terminals of connected loads for the duration of a 4-hour station blackout (SBO) event. The demands placed on the battery by an SBO event envelope the demands which would be placed on the batteries by any design basis event (DBE). The 250-VDC battery chargers are sized to charge the batteries while supplying the normal continuous DC loads. The Division 1 250-VDC battery system serves the Division 1 uninterruptible power supply (UPS), RCIC motor operated valves, RCIC turbine pumps and several other non-critical loads. The Division 2 250-VDC battery system supplies power for the HPCI motor operated valves, HPCI turbine oil pumps, the Division II control room heating, ventilation, and air conditioning DC control circuits and UPS.

Two 125-V battery systems are provided, each of which feeds separate DC buses. Each 125-VDC battery is sized to provide adequate voltage at the terminals of connected loads for the duration of a 4-hour SBO event and is capable of meeting power requirements during a DBE. The 125-VDC batteries provide control power for plant equipment such as the in-plant 13.8 kV AC breakers, 4160 VAC breakers, 480 VAC load center breakers, auxiliary control power for the 1R & 2R transformers, various control relays, annunciators, and some emergency lighting. Three chargers energized from different essential AC power sources are provided for the two batteries, one for each system and one as a common spare. Failure of any one charger will not prevent charging of either battery system. The common standby charger is in a room separate from the other two chargers and can be connected manually to either battery bus by operating breakers and switches locally at the chargers and battery panels. Each 125-VDC charger can carry the normal 125-VDC load and at the same time supply additional charging current to keep the batteries in a fully charged condition.

Technical Evaluation

The LAR has requested to use the RICT program to extend the existing CTs for several MNGP TS 3.8, “Electrical Power Systems,” conditions. Attachment 4 of the LAR identifies mark-ups of the proposed changes to MNGP TS 3.8.1, Conditions A, B, C, and D, TS 3.8.4, Conditions A and B, and TS 3.8.7, Conditions A and B.

The NRC staff reviewed information pertaining to the proposed electrical power systems TS conditions in the LAR, the MNGP USAR, and applicable TS LCO and TS Bases to verify the capability of the affected electrical power systems to perform their safety functions (assuming no additional failures) is maintained. To achieve that objective, the staff verified whether each proposed TS condition’s design success criteria reflect the redundant or absolute minimum electrical power source/subsystem required to be operable by the LCOs to support the safety functions necessary to mitigate postulated DBAs, safely shutdown the reactor, and maintain the reactor in a safe shutdown condition. The NRC staff further reviewed the remaining credited power source/equipment to verify whether the proposed condition satisfies its design success criteria. In conjunction with reviewing the remaining credited power source/equipment, the NRC staff considered supplemental electrical power sources/equipment (not necessarily required by the LCOs and can be either safety or non-safety related) that are/is available at MNGP and capable of performing the same safety function of the inoperable electrical power source/equipment. In addition, the NRC staff verified that the licensee provided examples of RMAs that are appropriate to monitor and control risk for applicable TS conditions.

Appendix E of the MNGP USAR discusses the conformance of the MNGP electrical power systems with the AEC draft GDC, Criteria 24 and 39. Criteria 24 and 39 are reflected, in part, in the electrical power systems TS LCOs, which ensure redundant electrical power sources/equipment are operable (in operating modes). When an Action is entered in TS 3.8 LCO due to an inoperable electrical power source or piece of equipment, the redundancy is not maintained. Therefore, the NRC staff finds that Criterion 39 is not met temporarily during the RICT program entry for the proposed TS 3.8 conditions since the redundancy is not maintained.

When an Action item for TS 3.8 LCO is entered due to an inoperable electrical power source or piece of equipment but both EDGs are maintained in the event of a loss of offsite power, Criterion 24 will still be met. If one EDG is inoperable in the event of a loss of all offsite power, the redundancy of the EDG required by the TS LCO (in operating modes) will not be maintained. In this case, the NRC staff finds that Criterion 24 will not be met temporarily during the RICT program entry for the proposed applicable TS 3.8 conditions since the redundancy is not maintained. The NRC staff also finds that operating the plant while remedial actions are being taken during the period the redundancy required by Criteria 24 and 39 and the LCOs is not maintained is allowed by 10 CFR 50.36(c)(2), which states: "When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met."

The LAR, Enclosure 1, Table E1-1, "In-scope TS/LCO Conditions to Corresponding PRA Functions," provides a listing of each TS 3.8 LCO condition to which the RICT program is proposed to be applied and information regarding the TSs such as SSCs and functions covered by the TS LCO conditions and applicable mode(s), and design success criteria. The NRC staff used the design success criteria to evaluate the DID of the electrical power systems during the application of the RICT program.

The LAR, Enclosure 1, Table E1-2, "In-Scope TS/LCO Conditions RICT Estimate," provides the RICT estimates for the proposed TS 3.8 conditions. The LAR Table E1-2 states "No Entry" for the RICT estimates for TS 3.8.1, Condition C, TS 3.8.4, Condition B, and TS 3.8.7, Conditions A and B. The licensee explained the "No Entry" in Note 1 to Table E1-2 by stating, "Several quantification results exceed the risk cap level of 1E-03 (CDF) or 1E-04 (LERF). Those LCOs are listed as "No Entry" given the quantified risk. However, it is possible that the LCO could be entered for a partial failure and would result in lower quantified risk. In a lower risk condition, entry into the RICT program would be allowed." The NRC staff verified that the LAR Table E1-2 RICT estimates for TS 3.8.1, Conditions A, B, and D, and TS 3.8.4, Condition A, are within 30 days as specified in the RICT program.

The LAR, Enclosure 12, "Risk Management Action Examples," describes the process for identification and implementation of RMAs applicable during extended CTs and provides examples of RMAs for one required offsite circuit inoperable, one EDG inoperable, one offsite circuit and one EDG inoperable, and Division 1 or 2 DC electrical power subsystem (battery charger) inoperable. The LAR stated that these example RMAs may be considered during the RICT program entry for a proposed TS 3.8 condition to reduce the risk impact and ensure adequate DID. The LAR also stated that plant procedures will provide guidance for the determination and implementation of RMAs when entering the RICT Program consistent with the guidance provided in NEI 06-09-A, Revision 0. The NRC staff verified that the above-mentioned example RMAs include the three categories of actions recommended by NEI 06-09-A, Revision 0, to control risk for maintenance activities.

The NRC staff's evaluation of the proposed changes considered several potential plant conditions allowed by the proposed RICTs. The NRC staff also considered the available redundant or diverse means to respond to various plant conditions. In these evaluations, the NRC staff examined the safety significance of different plant conditions resulting in both shorter and longer CTs. The plant conditions evaluated are discussed in more detail below.

TS 3.8.1 – AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and
b. Two emergency diesel generators (EDGs).

The AC sources to the Class 1E AC electrical power distribution system consist of three offsite power sources through transformers 2R, 1R, and 1AR in normal alignment and two onsite EDGs. Auxiliary power is supplied by the primary station auxiliary transformer 2R during normal power operation. Provisions are made for an automatic, fast transfer of the auxiliary load from the primary station auxiliary transformer, 2R, to the reserve transformer 1R. In the event the reserve transformer 1R is unable to accept load, the essential buses are automatically transferred to the reserve auxiliary transformer 1AR. These transformers supply power to the equipment used to maintain a safe plant.

The proposed RICT program will apply to TS 3.8.1, Conditions A, B, C, and D, as discussed earlier in Section 2.2 of this SE.

For Condition A (one required offsite circuit inoperable), the design success criterion in the LAR, Table E1-1 is “One qualified circuit to the grid for a Class 1E 4.16 kV essential bus.” During the RICT program entry for TS 3.8.1, Condition A, the remaining required offsite circuit and the onsite EDGs will be capable to safely shut down the reactor and maintain the MNGP unit in a safe shutdown condition in the event of a DBA or transient with offsite power available. Potentially, the third qualified offsite circuit will also be available as a power supply to the safe shutdown loads if it is not used to meet LCO 3.8.1.a (thereby, be able to exit Condition A). In the event of loss of all offsite power concurrent with the DBAs, as discussed in the MNGP USAR Section 14.7.2.2.5, “Effects of Unavailability of offsite power,” (ADAMS Accession No. ML20003D155), the onsite EDGs will be capable of supplying power to the minimum ESF systems required to mitigate the consequences of postulated DBAs.

For Condition B (one EDG inoperable), the design success criterion in the LAR, Table E1-1 is “One EDG.” During a RICT program entry for Condition B, the remaining EDG and two required offsite circuits will be capable of supplying power to the ESF systems required to mitigate DBAs or transients with offsite power available. In addition, the third offsite circuit through the reserve auxiliary transformer will be available to supply the ESF systems, if needed. In the event of loss of all offsite power concurrent with the DBAs, as discussed in the MNGP USAR Section 14.7.2.3.2, “GE/GNF Single Failure Considerations,” the remaining EDG will be capable of powering the minimum ESF systems required to mitigate the consequences of postulated DBAs.

For Condition C (two required offsite circuits inoperable), the design success criterion in the LAR, Table E1-1 is “One qualified circuit to the grid for a Class 1E 4.16 kV essential bus.” During a RICT program entry for Condition C, the remaining qualified offsite circuit (if it is available and not used to exit Condition C) and the onsite EDGs will be capable, as discussed in MNGP USAR Section 14.7.2.2.5, of supplying power to the ESF systems required to mitigate DBAs or transients.

For Condition D (one required offsite circuit inoperable and one EDG inoperable), the design success criterion in the LAR, Table E1-1 is “One qualified circuit to the grid and one EDG for a Class 1E 4.16 kV essential bus.” Condition D is essentially the combination of Conditions A and B. During a RICT program entry for Condition D, the remaining offsite circuits (i.e., the remaining required offsite circuit and the third qualified offsite circuit, if available) and the onsite EDG will be capable of supplying power to the ESF systems required to mitigate DBAs or transients with offsite power available. In the event of loss of all offsite power concurrent with the DBAs, as discussed in the MNGP USAR Section 14.7.2.3.2, the remaining EDG will be capable of supplying power to the minimum ESF systems required to mitigate the consequences of postulated DBAs.

Based on the above discussion, the NRC staff finds that during the RICT program entry for TS 3.8.1, Conditions A, B, C, and D, the DID of the electrical power systems that ensures onsite AC power to key safety-related equipment required to operate during DBAs with or without offsite power is reduced to at least the required minimum electrical power source (i.e., one EDG to support one train of required ESF equipment). Based on the availability of at least one train of onsite electrical power system to support the safety functions, the NRC staff finds the proposed change to MNGP TS 3.8.1 acceptable.

<u>TS 3.8.4 – DC Sources – Operating</u>

LCO 3.8.4 The Division 1 and Division 2 125 VDC and 250 VDC electrical power subsystems shall be OPERABLE.

Each division includes one 250 VDC battery system and one 125 VDC battery system. Each 250 VDC battery system includes a 125 VDC spare charger. The design of the 125 VDC electrical power system includes a common standby 125 VDC spare charger that may be shared between the Division 1 and Division 2 125 VDC electrical power subsystems.

The LCO 3.8.4 requires: (1) each Division 1 and Division 2 250 VDC subsystem, consisting of two 125 V batteries in series, two battery chargers, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus; and (2) each Division 1 and Division 2 125 VDC subsystem consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus to be operable.

The proposed RICT program will apply to TS 3.8.4, Conditions A and B, as discussed earlier in Section 2.2 of this SE.

For Condition A (one or more required battery chargers on Division 1 or Division 2 inoperable), the design success criterion in the LAR, Table E1-1 is two battery chargers for each 250 VDC electrical power subsystem and one battery charger for each 125 VDC electrical power subsystem. The worst-case Condition A is one division with all required chargers (two battery chargers in the 250 VDC electrical power subsystem and one battery charger in the 125 VDC electrical power subsystem) inoperable. During the RICT program entry for the worst-case Condition A, the remaining battery chargers (in the 250 VDC and 125 VDC electrical power subsystems) in the unaffected division and the spare chargers will be capable of supplying power to the DC loads required to safely shutdown the reactor and maintain the MNGP unit in a safe shutdown condition in the event of a DBA or transient with or without offsite power. The batteries in the redundant division without the chargers will be available for a short duration to support safe shutdown of the unit.

For Condition B (one Division 1 or Division 2 DC electrical power subsystem inoperable for reasons other than Condition A), the design success criterion in the LAR, Table E1-1 is “One of two electrical power subsystems.” Condition B 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. The worst-case Condition B is one division with inoperable battery charger(s) and associated inoperable battery(ies). During the RICT program entry for the worst-case Condition B, the remaining division of DC electrical power subsystem (battery chargers and batteries for the redundant 250 VDC and 125 VDC electrical power subsystems) will be capable of supplying power to the DC loads required to safely shutdown the reactor and maintain the MNGP unit in a safe shutdown condition in the event of a DBA or transient with or without offsite power.

Based on the above discussion, the NRC staff finds that, during a RICT program entry for TS 3.8.4, Conditions A and B, the DID of the electrical power systems that ensures DC power to key safety-related equipment required to operate during DBAs with or without offsite power is reduced to at least one division DC electrical power system (chargers and batteries). The NRC staff finds the proposed change to MNGP TS 3.8.4 acceptable because one division of the DC electrical power system is available to support the safety functions.

TS 3.8.7 – Distribution Systems – Operating

LCO 3.8.7 Division 1 and Division 2 AC and DC electrical power distribution subsystems shall be OPERABLE.

As described in the TS Bases and in the USAR Sections 8.5 and 8.7, the onsite Class 1E AC and DC electrical power distribution system is divided into redundant and independent AC and DC electrical power distribution subsystems.

The primary AC electrical power distribution subsystem for each division consists of a 4.16-kV essential bus (essential bus 15 for Division 1 and essential bus 16 for Division 2) and one 480-VAC load center (load center 103 for Division 1 and load center 104 for Division 2). Each load center is supplied from the associated 4.16-kV essential bus via a transformer.

There are two independent 125/250 VDC electrical power distribution subsystems, one per division, that support the necessary power for ESF functions, each consisting of a 125/250 VDC distribution panel (distribution panel D31 for Division 1 and distribution panel D100 for Division 2). There are two independent 125 VDC electrical power distribution subsystems, one per division, that support the necessary power for safety functions, each consisting of a 125 VDC distribution panel (distribution panel D11 for Division 1 and distribution panel D21 for Division 2).

The proposed RICT program will apply to TS 3.8.7, Conditions A and B, as discussed earlier in Section 2.2 of this SE.

For Condition A, the design success criterion in the LAR, Table E1-1 is "One AC electrical power distribution subsystem capable of supporting minimum safety functions." With one or more required AC buses or load centers inoperable without loss of function, the Condition A worst scenario is one division without AC power. During the RICT program entry for the worst-case Condition A, the remaining AC electrical power distribution subsystems in the unaffected division can support the minimum safety functions necessary to safely shut down the reactor and maintain it in a safe shutdown condition in the event of a DBA or transient with or without offsite power.

For Condition B, the design success criterion in the LAR, Table E1-1 is "One DC electrical power distribution subsystem capable of supporting minimum safety functions." Each division includes a 250-VDC and a 125-VDC electrical power distribution subsystem. With one or more DC distribution subsystem(s) inoperable without loss of function, the Condition B worst-case scenario is one division without adequate DC power (both the charger and associated battery inoperable). During the RICT program entry for the worst-case Condition B, the remaining DC electrical power distribution subsystems in the unaffected division can support the minimum safety functions necessary to safely shut down the reactor and maintain it in a safe shutdown condition in the event of a DBA or transient with or without offsite power.

According to TSTF-505, Revision 2, the RICT program can only be used when there is no TS loss of function. The NRC staff noted that in some cases, Condition A or Condition B would be a loss of function condition when both 4.16 kV essential buses in both divisions (Condition A) or both 125/250-VDC distribution panels in both divisions (Condition B) were inoperable. However, the proposed alternate RICTs for Condition A or Condition B do not exclude the cases for loss of function of Condition A or Condition B from the RICT program. Thus, the NRC staff requested the licensee discuss how the MNGP design-basis functions are met without the exclusion of loss of safety function conditions in the proposed RICT program for Conditions A and B.

The response to the NRC staff's request (RAI 21) provided by letter dated December 21, 2020, stated that the proposed RICT program for Condition A or Condition B does not apply to a loss of function because TS 3.8.7 Condition D, "two or more electrical power distribution subsystems inoperable that result in a loss of function," is required to be entered upon a loss of function. The licensee further stated that the loss of function in Condition D could be due to, "both divisions of AC, both divisions of DC, or one division of AC with the opposite division of DC."

The NRC staff notes that since the loss of function due to either both divisions of AC electrical power distribution subsystems inoperable (Condition A) or both divisions of DC electrical power distribution subsystems inoperable (Condition B) requires entry into Condition D, the MNGP RICT program does not apply to Condition A or B when a loss of function occurs.

Based on the above discussion, the NRC staff finds that, during a RICT program entry for TS 3.8.7, Conditions A and B, the DID of the electrical power distribution systems that ensures AC and DC power, respectively, to key safety-related equipment required to operate during DBAs with or without offsite power is reduced to at least one division of AC or DC electrical power distribution subsystems. The NRC staff finds the proposed change to MNGP TS 3.8.7 acceptable because one division of AC or DC electrical power distribution subsystem is available to support the safety functions.

Evaluation of Electrical Power Systems Conclusion

The NRC staff finds that while the redundancy is not maintained (e.g., one train of a two-train system is inoperable), the CT extensions in accordance with the RICT program are acceptable because: (1) the capability of the systems to perform their safety functions (assuming no additional failures) is maintained, and (2) the licensee's demonstration of identifying and implementing compensatory measures or RMAs, in accordance with the RICT program, are appropriate to monitor and control risk.

3.1.2.3 Evaluation of Instrumentation and Control (I&C) Systems

The LAR requested to use the RICT program to extend the existing CT for the TS conditions discussed in this section. The NRC staff's evaluation of the proposed changes considered a number of potential plant conditions allowed by the new TSs and considered what redundant or diverse means were available to assist the licensee in responding to various plant conditions. The NRC staff followed the guidance in the RG 1.174 and further elaborated in the RG 1.177 to assess the proposed changes' consistency with DID criteria. The applicable DID criteria to the affected I&C systems are:

1. Over-reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided.

2. System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system (e.g., there are no risk outliers).

3. Defenses against potential CCFs are maintained and the potential for the introduction of new CCF mechanisms is assessed.

4. The intent of the plant's design criteria is maintained.

The LAR confirmed that the proposed changes do not alter MNGP I&C system designs. Consequently, the NRC staff concludes that the proposed changes do not alter the ways in which the MNGP I&C systems fail, do not introduce new CCF modes, and the system independence is maintained. The NRC staff finds that some proposed changes reduce the level of redundancy of the affected I&C systems during the proposed RICT period and this reduction may reduce the level of defense against some CCFs; however, the NRC staff finds, as described below, such reduction in redundancy and defense against CCFs during the RICT period are acceptable due to existing diverse means available to maintain adequate DID against a potential single failure during the RICT for the MNGP I&C systems.

The following summarizes the NRC staff's evaluation with respect to the DID principle for the functions identified in the LAR by identifying associated diverse means that maintain adequate DID against potential single failure during the RICT for the MNGP I&C systems. The proposed changes to the LCOs in this section are discussed in Section 2.2 of this SE.

<u>TS 3.3.1.1 – RPS Instrumentation</u>

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

The RPS, as described in MNGP USAR, Section 7.6.1.2.1 (ADAMS Accession No. ML20003D151), and Table E1-3 of the LAR, includes sensors, relays, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. There are at least four redundant sensor input signals from each of these parameters (except for the reactor mode switch in shutdown and manual scram signals). Some channels include electronic equipment (e.g., trip units) that compares measured input signals with preestablished setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an RPS trip signal to the trip logic.

These changes are applicable to all safety functions listed in MNGP TS 3.3.1.1-1, except to those specified in Required Actions A.2 NOTE and B NOTE. The LAR described the redundancy design features of these functions in Table E1-3, Enclosure 1.

Under Condition A with one or more required channels inoperable, the applicable safety functions listed in the MNGP TS 3.3.1.1-1 might lose their RPS trip capabilities. If this is the case, then TS LCO 3.3.1.1 Condition C is entered, which is not risk-informed, to restore its trip capability within 1 hour. Therefore, this configuration assures that the risk-informed Actions A.1 and A.2 shall only apply to the non-loss of function conditions.

The letter dated December 21, 2020 confirmed in Table 24-1 of the RAI response, and the NRC staff verified, that during the extended CTs at least one diverse means is available, including manual actuations, for all functions listed in MNGP TS 3.3.1.1-1, Actions A.1 and A.2, except Functions 2.a, 2.b, 2.c, 2.d, 2.f or 2.g. for Action A.2, for every MNGP USAR Chapter 14 DBA for which these affected functions are credited.

The RAI response confirmed, and the NRC staff verified, that these "manual actuations" are defined in plant operation procedures to which operators are trained. Selected manual actions are modeled in the PRA when it is determined that credit for the actions has a positive risk impact. Those actions that are modeled incorporate appropriate human error probabilities. The NRC staff concludes this approach is consistent with the "not over-relying on programmatic activities as compensatory measures" principle provided in RG1.174 Revision 2 and is acceptable.

Under Condition B, "one or more Functions with one or more required channels inoperable," the applicable safety functions listed in the MNGP TS 3.3.1.1-1 might lose their RPS trip capabilities. If this is the case, then TS LCO 3.3.1.1 Condition C is entered, which is not risk-informed, to restore its trip capability within 1 hour. Therefore, this configuration assures that the risk-informed ACTION B.1 and B.2 shall only apply to the non-loss of function conditions.

The letter dated December 21, 2020 confirmed in Table 24-1 of the RAI response, and the NRC staff verified that during the extended CTs at least one diverse means is available, including manual actuations, for all functions listed in MNGP TS 3.3.1.1-1, Actions B.1 and B.2, except Functions 2.a, 2.b, 2.c, 2.d, 2.f or 2.g. for Actions B.1 and B.2, for every MNGP USAR Chapter 14 DBA for which these affected functions are credited.

The RAI response confirmed, and the NRC staff verified that these "manual actuations" are defined in plant operation procedures to which operators are trained. Selected manual actions are modeled in the PRA when it is determined that credit for the actions has a positive risk impact. Those actions that are modeled incorporate appropriate human error probabilities. The NRC staff concludes this approach is consistent with the "not over-relying on programmatic activities as compensatory measures" principle suggested in the RG1.174, Revision 2, and is acceptable.

TS 3.3.2.2 – Feedwater Pump and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Four channels of Feedwater Pump and Main Turbine High Water Level Trip Instrumentation shall be OPERABLE.

The feedwater pump and main turbine high water level trip instrumentation, as described in the MNGP USAR, Section 7.7.4, and Table E1-4 of the LAR, has four channels of Reactor Vessel Water Level – High instrumentation arranged such that one specific Reactor Vessel Water Level – High channel in each of two trip systems or both channels in a trip system cause the trip function (1 out of 2 taken twice logic).

Under Condition A with one or more required channels inoperable, this safety function might lose its trip capability. If this is the case, then the plant operation enters TS LCO 3.3.2.2

Condition B, which is not risk-informed, to restore its trip capability within 2 hours. Therefore, this configuration assures that the risk-informed ACTION A.1 shall only apply to the non-loss of function conditions.

The letter dated December 21, 2020, confirmed in the Table 24-2 of the RAI response, and the NRC staff verified that during the extended CTs, the feedwater pump and main turbine high water level trip function, for every MNGP USAR Chapter 14 DBA for which this function is credited there is at least one diverse means available other than the manual actuation.

<u>TS 3.3.4.1 – Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation</u>

LCO 3.3.4.1 Two channels per trip system for each ATWS-RPT instrumentation function listed below shall be OPERABLE:

a. Reactor Vessel Water Level – Low; and
b. Reactor Vessel Steam Dome Pressure – High.

The ATWS-RPT Instrumentation, as described in the MNGP USAR, Section 7.6.2.2, and Table E1-5 of the LAR, consists of two independent trip systems, with two channels of Reactor Vessel Steam Dome Pressure – High and two channels of Reactor Vessel Water Level – Low in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each function. Either two Reactor Vessel Water Level – Low Low or two Reactor Vessel Steam Dome Pressure – High signals are needed to trip a trip system. The outputs of the channels in a trip system are combined in a logic arrangement such that either trip system will trip both recirculation pumps.

Under Condition A with one or more required channels inoperable, either of the two applicable safety functions might lose its trip capability. If this is the case, then TS 3.3.4.1 Condition B is entered, which is not risk-informed, to restore its trip capability within 72 hours. If both functions lose their trip capabilities, then TS 3.3.4.1 Condition C is entered, which is not risk-informed, to restore its trip capability within 1 hour. Therefore, the proposed changes assure that the risk-informed Action A.1 shall only apply to the non-loss of function conditions.

The December 21, 2020, letter confirmed in the Table 24-3 of the RAI, and the NRC staff verified that during the extended CTs, the ATWS-RPT Instrumentation, for every MNGP USAR Chapter 14 DBA that these affected functions are credited there is at least one diverse means available other than the manual actuation.

<u>TS 3.3.5.1 – ECCS Instrumentation</u>

LCO 3.3.5.1 The ECCS Instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

The ECCS instrumentation, as described in the MNGP USAR, Section 7.1.1.2, and Table E1-6 of the LAR, actuates CS, low-pressure cooling injection (LPCI), HPCI, ADS, and the EDGs. The ECCS instrumentation also employs diversity in the number and variety of different inputs which will actuate the associated equipment.

Under Condition A with one or more required channels inoperable, some ECCS functions listed in MNGP TS 3.3.5.1-1 could lose their initiation capabilities. The application of the RICT program to these loss of function conditions shall be prohibited per notes specified in the RICT program which state that “[n]ot applicable when a loss of function occurs” and “Risk Informed Completion Time Program not applicable to loss of function.”

The December 21, 2020, letter confirmed in the Table 24-4 of the RAI, and the NRC staff verified that during the extended CTs, the ECCS Instrumentation, for every MNGP USAR Chapter 14 DBA that these affected Functions are credited for, there are at least one diverse means available other than the manual actuation.

The licensee also confirmed that the ECCS Function 3.c, HPCI System, Reactor Vessel Water Level – High, is not credited by the MNGP USAR Chapter 14 DBA analysis. Its diversity analysis is not required by this RICT SE analysis.

<u>TS 3.3.5.2 – Reactor Core Isolation Cooling (RCIC) System Instrumentation</u>

LCO 3.3.5.2 The RCIC System instrumentation for each Function in Table 3.3.5.2-1 shall be OPERABLE.

The RCIC instrumentation, as described in the MNGP USAR, Section 7.6.3, and Table E1-7 in the LAR, initiates actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (the main condenser) and normal coolant makeup flow from the reactor feedwater system is unavailable, such that injection by the low pressure ECCS pumps does not occur.

Under Condition A with one or more required channels inoperable, one or more RCIC functions listed in MNGP TS Table 3.3.5.2-1 might lose initiation capabilities. The application of the RICT program to these loss of function conditions shall be prohibited per the note specified in the RICT program which states that “[n]ot applicable when a loss of function occurs.”

The December 21, 2020, letter confirmed in the Table 24-5 of the RAI response, and the NRC staff verified, that there are diverse means available, other than the manual actuation, during the extended CTs for every MNGP USAR Chapter 14 DBA that these affected functions are credited for.

<u>TS 3.3.6.1 – Primary Containment Isolation Instrumentation</u>

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

The Primary Containment Isolation instrumentation, as described in the MNGP USAR, Section 7.6.3, and Table E1-8 of the LAR, initiates primary containment and reactor coolant pressure boundary isolation. The input parameters to the isolation logics are (a) reactor vessel water level, (b) area ambient temperatures, (c) main steam line (MSL) flow measurement, (d) SLC System Initiation (HPCI and RCIC steam line flow, (g) drywell pressure, (h) HPCI and RCIC steam line pressure, (i) reactor water cleanup flow (RWCU), and (j) reactor steam dome pressure.

Under Condition A with one or more required channels inoperable, functions listed in Table 3.3.6.1-1 might lose capabilities to initiate the primary containment isolation. If this is the case, then TS LCO 3.3.6.1 Condition B, “One or more Functions with primary containment isolation capability not maintained” and Action B.1, which is not risk informed, is entered to restore the primary containment isolation capability within 1 hour. Therefore, the proposed changes shall not apply to the RICT program under these loss of function conditions.

The December 21, 2020, letter confirmed in the Table 24-6 of the RAI response, and the NRC staff verified that all primary containment isolation instrumentation credited in the MNGP USAR Chapter 14 DBA analyses (except Function 5.d – RWCU system isolation, SLC system initiation) has diverse means available during the extended CT, other than manual actuation. The only diverse means for Function 5.d is manual actuation.

The licensee confirmed, and the NRC staff verified that all “manual actuations” specified in the LAR are defined in plant operation procedures to which operators are trained. Selected manual actions are modeled in the PRA when it is determined that credit for the actions has a positive risk impact. Those actions that are modeled incorporate appropriate human error probabilities. The NRC staff concludes this approach is consistent with the “not over-relying on programmatic activities as compensatory measures” principle suggested in the RG 1.174, Revision 2, and is acceptable.

<u>TS 3.3.7.2 – Mechanical Vacuum Pump Isolation Instrumentation</u>

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

The mechanical vacuum pump isolation instrumentation, as described in the MNGP USAR, Section 14.7.1, and Table E1-9 of the LAR, initiates a trip of the mechanical vacuum pump and isolation of the isolation valves following events in which main steam radiation monitors exceed a predetermined value. Tripping and isolating the mechanical vacuum pump limits control room and offsite doses in the event of a control rod drop accident (CRDA).

The isolation logic consists of two independent trip systems, with two channels of the Main Steam Line Tunnel Radiation - High Function in each trip system. The outputs from two channels provide input into one trip system and the other two channels provide input into the other trip system. One channel must trip to trip a trip system and both trip systems must trip to initiate the mechanical vacuum pump isolation function (i.e., one-out-of-two taken twice logic arrangement). There is one mechanical vacuum pump breaker and two mechanical vacuum pump isolation valves associated with this function.

Under Condition A with one or more required channels inoperable, the Main Steam Line Tunnel Radiation - High Function might lose the capability to initiate the mechanical vacuum pump isolation. If this is the case, then TS LCO 3.3.7.2 Condition B, “Mechanical vacuum pump isolation capability not maintained” and Action B.1, which is not risk informed, is entered to restore the mechanical vacuum pump isolation capability within 1 hour. Therefore, the proposed changes shall not apply to the RICT program under these loss of function conditions.

The December 21, 2020, letter confirmed in the Table 24-7 of the RAI response, and the NRC staff verified that the mechanical vacuum pump isolation instrumentation credited in the MNGP USAR Chapter 14 DBA analyses has diverse means of actuation other than manual actuation during the extended CTs.

TS 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.

The LOP instrumentation, as described in the MNGP USAR, Section 8.4.1.3, and Table E1-10 in the LAR, monitors the 4.16 kV essential buses. If the monitors determine that insufficient offsite power is available, the buses are disconnected from the offsite power sources and connected to the onsite EDG power sources.

Each 4.16 kV essential bus has its own independent LOP instrumentation and associated trip logic. The voltage for each bus is monitored at two levels, which can be considered as two different undervoltage functions: 4.16 kV essential bus loss of voltage and 4.16 kV essential bus degraded voltage. Each function causes various bus transfers and disconnects. The 4.16 kV essential bus loss of voltage function is monitored by four undervoltage relays for each emergency bus, whose outputs are arranged in a one-out-of-two taken twice logic configuration (i.e., one channel in each of two trip systems must trip for LOP actuation). The 4.16 kV essential bus degraded voltage function is monitored by three undervoltage relays (with its associated time delay) for each emergency bus whose outputs are arranged in a two-out-of-three logic configuration.

Under Condition A with one or more required channels inoperable, functions listed in Table 3.3.8.1-1 of the TS lose their trip capabilities. The application of the RICT program to these loss of function conditions shall be prohibited per the note specified in the RICT program which states that "[n]ot applicable when a loss of function occurs." In addition, TS LCO 3.3.8.1 Condition B, "Required Action and associated Completion Time not met" and ACTION B.1, which is not risk informed, is entered to immediately declare associated EDG inoperable. Therefore, the proposed changes shall not apply to the RICT program under these loss of function conditions.

The December 21, 2020, letter confirmed in the Table 24-8 of the RAI response and the NRC staff verified, that during the extended CTs there are diverse means available in addition to manual actuation for the LOP instrumentation for every MNGP USAR Chapter 14 DBA for which these affected functions are credited.

Evaluation of Instrumentation and Control Conclusion

Since the licensee did not propose any changes to the design basis, the independency and the fail-safe principle remain unchanged. The LAR stated that the proposed changes did not include any TS loss of function conditions. However, it is recognized that while in an Action statement, redundancy of the given protective feature will be temporarily reduced, and, accordingly, the system reliability will be reduced. The LAR stated in the description of proposed changes to the I&C that at least one redundant or diverse means (e.g., other automatic features or manual action) to accomplish the safety functions (e.g., reactor trip, safety injection, or containment isolation) remains available during the use of the RICT. The NRC staff reviewed the licensee's proposed TS changes to assess the availability of the redundant or diverse means to accomplish the safety function(s). The NRC staff finds that the availability of the redundant or diverse protective features provide sufficient DID to accomplish the safety functions, allowing for the extension of CTs in accordance with the RICT program. The NRC staff finds that the licensee proposed RICT program to the identified I&C systems is in compliance with 10 CFR 50.36(b) and 10 CFR 50.55a(h).

The NRC staff reviewed the licensee's proposed TS changes and supporting documentation. The NRC staff finds that while the instrumentation and control redundancy is reduced, the CT extensions implemented in accordance with the RICT program are acceptable because: (1) the capability of the instrumentation and control systems to perform their safety functions is maintained, (2) redundant or diverse means to accomplish the safety functions exist, and (3) the licensee will identify and implement risk management actions to monitor and control risk in accordance with the RICT program.

3.1.2.4 Key Principle 2 Conclusions

The LAR proposes to modify the TS requirements to permit extending selected CTs using the RICT program in accordance with NEI 06-09-A. The NRC staff finds that extending the selected CTs with the RICT program following loss of redundancy, but maintaining the capability of the system to perform its safety function, is an acceptable reduction in DID provided that the licensee identifies and implements RMAs in accordance with the RICT program during the extended CT.

Quantitative risk analysis, qualitative considerations including compensatory measures, and retaining the current CT for loss of all trains of a required system, assure that DID is maintained to assure adequate protection of public health and safety. The NRC staff finds that the proposed changes are consistent with the DID philosophy because:

- System redundancy (with the exceptions discussed above), independence, and diversity commensurate with the expected frequency and consequences of challenges to the system is preserved.
- Adequate capability of design features without an overreliance on programmatic activities as compensatory measures is preserved.
- The intent of the plant's design criteria continues to be met.

Therefore, the NRC staff finds that this proposed change meets the second key safety principle of RG 1.177 and is, therefore, acceptable. Additionally, the NRC staff concludes that the proposed changes are consistent with the DID philosophy as described in RG 1.174.

3.1.3 Key Principle 3: Evaluation of Safety Margins

Section 2.2.2 of RG 1.177, Revision 1, states, in part, that sufficient safety margins are maintained when:

- Codes and standards … or alternatives approved for use by the NRC are met.
- Safety analysis acceptance criteria in the final safety analysis report (FSAR) are met or proposed revisions provide sufficient margin to account for analysis and data uncertainties.

The licensee is not proposing in this application to change any quality standard, material, or operating specification. In the LAR, the licensee proposed to add a new program, "Risk Informed Completion Time Program," in Section 5.5.16, "Administrative Controls," of the TSs, which would require adherence to NEI 06–09-A.

The NRC staff evaluated the effect on safety margins when the RICT is applied to extend the CT up to a backstop of 30 days in a TS condition with sufficient trains remaining operable to fulfill the TS safety function. Although the licensee will be able to have design basis equipment out of service longer than the current TS allow, any increase in unavailability is expected to be insignificant and is addressed by the consideration of the single failure criterion in the design-basis analyses. Acceptance criteria for operability of equipment are not changed and use of the RICT only when the system(s) retain(s) the capability to perform the applicable safety function(s) ensures that the current safety margins are retained. Safety margins are also maintained if PRA functionality is determined for the inoperable train, which would result in an increased CT. Credit for PRA functionality, as described in NEI 06-09-A, is limited to the inoperable train, loop, or component. The reduced but available functionality may support a further increase in the CT consistent with available safety margin. The specified safety function is still being met by the operable train and therefore requires no evaluation of PRA functionality to meet the design basis success criteria.

3.1.3.1 Key Principle 3: Conclusions

The NRC staff finds that the design-basis analyses for MNGP remain applicable. Although the licensee will be able to have design-basis equipment out of service longer than the current TS allow and the likelihood of successful fulfillment of the function will be decreased when redundant train(s) are not available, the capability to fulfill the function will be retained when the available equipment functions as designed. Any increase in unavailability because less equipment is available for a longer time is included in the RICT evaluation. Therefore, safety margins are not affected adversely by the implementation of the RICT program. The NRC staff concludes that the proposed change meets the third key safety principle of RG 1.177 and is acceptable.

3.1.4 Key Principle 4: Change in Risk Consistent with the Safety Goal Policy Statement

TS Section 5.5.16 "Risk Informed Completion Time Program," states that the RICT "must be implemented in accordance with NEI 06-09, 'Risk-Informed TSs Initiative 4b: RMTS Guidelines,' Revision 0-A, November 2006." NEI 06-09-A is a methodology for a licensee to evaluate and manage the risk impact of extensions to TS CTs. Permanent changes to the fixed TS CTs are typically evaluated by using the three-tiered approach described in Chapter 16.1 of the SRP, RG 1.177, and RG 1.174. This approach addresses the calculated change in risk as measured by the change in ΔCDF and ΔLERF, as well as the ICCDP and ICLERP; the use of compensatory measures to reduce risk; and the implementation of a CRMP to identify risk-significant plant configurations.

The NRC staff evaluated the licensee's processes and methodologies for determining that the change in risk from implementation of RICTs will be small and consistent with the intent of the Commission's Safety Goal Policy Statement (published in the *Federal Register* 51 FR 30028 (August 21, 1986)), as discussed below. The NRC staff evaluated the licensee's proposed changes against the three-tiered approach in RG 1.177, Revision 1, for the licensee's evaluation of the risk associated with a proposed TS CT change. The results of the staff's review are discussed below.

3.1.4.1 Tier 1: PRA Capability and Insights

The first tier evaluates the impact of the proposed changes on plant operational risk. The Tier 1 review involves two aspects: (1) the technical acceptability of the PRA models and their application to the proposed changes, and (2) a review of the PRA results and insights described in the licensee's application.

PRA Technical Acceptability

RG 1.174 states that the scope, level of detail, and technical adequacy of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process. The NRC's SE as described in NEI 06-09 0-A states that the PRA models should conform to the guidance in RG 1.200, Revision 1 (ADAMS Accession No. ML070240001). The current version is RG 1.200, Revision 2, which clarifies that the current applicable PRA standard is ASME/ANS RA-Sa–2009, "Addenda to ASME RA-S–2008, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications."

The NRC staff evaluated the PRA acceptability information provided in Enclosure 2 of the LAR and in the supplements dated December 21, 2020, and April 20, 2021. This included industry peer review results and the licensee's self-assessment of the PRA models (for internal events including internal flooding and for fire) against the guidance in RG 1.200, Revision 2. The licensee screened out all external hazard events, except for seismic, as insignificant contributors to RICT calculations. The PRA model with modifications is used as the CRMP model.

As stated in LAR Section 3 of Enclosure 2, FLEX equipment actions are credited in the PRA model. The LAR further states that since FLEX mitigation strategies have minimal impact on CDF and LERF values it would have minimal impact on RICT calculations. The NRC staff has previously noted concerns crediting FLEX in risk-informed applications, such as the fact that no industry approved data exist for FLEX equipment and the uncertainty of assessing FLEX human actions. Therefore, the NRC staff requested additional information. In response, the licensee explained credit is limited to two FLEX fuel oil transfer cubes. The licensee provided a sensitivity study that demonstrated that the credit of FLEX resulted in no more than a three percent impact on any of the proposed RICT TS LCOs. Based on these results, the NRC staff finds that the sources of uncertainty associated with FLEX do not impact this application and that the inclusion of FLEX is acceptable for the application.

Internal Events PRA (Including Internal Flooding)

The NRC staff review of the MNGP internal events (including internal flooding) PRA was based on the results of a full-scope peer review of the internal events PRA and a facts and observations (F&Os) closure review described in LAR Enclosure 2. The full-scope peer review was performed in April 2013 based on the process described in NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 3 (ADAMS Accession No. ML083430462) using the PRA standard and RG 1.200, Revision 2, to Capability Category (CC) II supporting requirements. The F&O closure review was performed in October 2017 against the same criteria on supporting requirements with finding-level F&Os from the 2013 full-scope review. The F&O closure review was performed by an independent assessment team consistent with guidance in Appendix X of NEI 05-04 and clarifications in the NRC's acceptance letter dated May 3, 2017 (ADAMS Accession No. ML17079A427). Based on the F&O closure review, all open F&Os were closed. LAR Enclosure 2 of the LAR states that the resolutions to close out the F&Os were determined not to constitute PRA upgrades as defined by the PRA standard. However, the LAR continued by stating that the PRA model has subsequently been updated. The NRC staff requested confirmation that none of the PRA model updates constituted a PRA upgrade. In response, the licensee provided a list of model updates since April 2013 that identified only one that qualified as an PRA upgrade, related to incorporation of a convolution analysis to estimate the probability of AC power recovery. The licensee stated it was subsequently peer reviewed and that no F&Os were identified. The NRC staff reviewed the other updates and determined that they were appropriately treated as model maintenance.

During its audit, the NRC staff reviewed the internal events F&O closure report by the independent assessment teams. NRC staff found that the independent assessment team (IAT) team reviewed the open finding-level F&Os to supporting requirements applicable to the F&O at CC II. The NRC staff team found that the licensee performed a self-assessment of whether the resolution of a finding could constitute an upgrade of the PRA as defined by the PRA standard and the NRC staff found that these determinations and their bases were reviewed by the IAT.

The LAR in Table A5-1, as revised in April 20, 2020 supplement, included implementation items that are required to be completed prior to the implementation of the RICT program. Table A5-1 states that prior to implementation of the RICT program, these items will be modeled in sufficient detail to accurately calculate a RICT: reactor protection system instrumentation (implementation item 1), mechanical vacuum pump system and isolation instrumentation (implementation item 2), and the automatic depressurization system and instrumentation (implementation item 3). This will facilitate performing a RICT calculation for TS LCO 3.3.1.1.A associated with reactor protection system instrumentation, TS LCO 3.3.5.1 G associated with emergency core cooling system instrumentation, and TS LCO 3.3.7.2 A associated with mechanical vacuum pump isolation instrumentation. Given the direct impact on these three RICT calculations, the NRC requested in RAI 05 an explanation of how these changes will meet CC II requirements of the PRA standard. In response, the licensee confirmed that all of the channels related to these three systems will be modeled in accordance with the CC II requirements of the PRA standard. The NRC staff has determined the scope and detail of the model additions are appropriate for this application.

Based on its review, the NRC staff finds that the internal events PRA, including internal flooding, has been adequately peer reviewed against the PRA standard and RG 1.200 and that the licensee has adequately closed the F&Os and will appropriately update the internal events PRA system modeling (implementation items 1, 2, and 3) prior to implementation of the RICT program; therefore, after completion of the implementation item listed, the internal events PRA, including internal flooding, is technically acceptable to support the RICT program.

Fire PRA

The NRC staff review of the MNGP fire PRA was based on the results of a full-scope peer review, an F&O closure review, a focused-scope peer review, and a second F&O closure review described in LAR Enclosure 2. The full-scope peer review of the fire PRA was performed in March 2015, using the NEI 07-12 process and the guidance in the PRA standard and RG 1.200, Revision 2. In December 2016, a focused-scope review was conducted to address the use of enhanced modeling methods related to heat soak and resulted in additional F&Os. The first MNGP fire PRA F&O closure review was performed on all the fire PRA findings in October 2017. The October 2017, F&O closure process for the MNGP fire PRA was performed consistent with guidance in Appendix X of NEI 07-12 and clarifications in the NRC's acceptance letter dated May 3, 2017. The second MNGP fire PRA F&O closure review was performed in April 2019, on the open finding-level F&Os and all previous F&Os were subsequently closed by the IAT. However, the IAT determined one model change constituted a PRA upgrade. A focused-scope peer review was performed by the IAT concurrent with the April 2019, F&O closure review on the fire scenario selection (FSS) technical element of the PRA standard and determined there was one resultant F&O.

During the audit, NRC staff reviewed the F&O closure reports by the IAT. NRC staff found that the IAT reviewed the open F&Os to PRA supporting requirements applicable to F&O at CC II. The NRC staff also found that the licensee performed a self-assessment of whether the resolution of a finding could constitute an upgrade of the PRA as defined by the PRA standard and found that these determinations and their bases were reviewed by the IAT.

LAR Enclosure 2, Table E2-1, presents the disposition for the one open F&O. The LAR stated that several recalculations and new sensitivity studies were performed and confirmed that the finding is not expected to have any impact on RICT calculations.

The NRC staff also requested additional information on the fire PRA modeling treatments that are not related to F&Os but have been previously identified as potential key assumptions and sources of uncertainty for fire PRAs and are discussed in the following paragraphs.

The NRC requested information about use of fire PRA methods that deviate from guidance provided in NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," (ADAMS Accession No. ML15167A401) or other acceptable guidance (e.g., frequently asked questions, NUREGs, or interim guidance documents). In response, the licensee stated the determination of the main control board ignition frequencies did not follow the guidance of Appendix L of NUREG/CR-6850, which resulted in higher initiation frequencies. Instead, the licensee applied a methodology previously reviewed by the staff (see letter dated August 8, 2017, ADAMS Accession No. ML17163A027), and determined to develop scenarios that are more detailed than those obtained using the guidance of Appendix L. The NRC staff determined that the licensee's approach for determining the main control board ignition frequencies is acceptable for this application because it relied upon previously approved methods.

The NRC requested information about use of reduced transient fire heat release rates (HRRs) below those prescribed in NUREG/CR-6850 and justification if reduced HRRs were used. The requested information included (1) identification of fire areas where a reduced transient fire HRR is credited and the reduced HRR value that was applied, (2) discussion of administrative controls that support justification for using the reduced HRR, (3) discussion of the required controls for ignition sources in these locations and the types and quantities of combustible materials needed to perform maintenance, (4) discussion of personnel traffic that would be expected through each location, and (5) results of any review of records related to compliance with the transient combustible and hot work controls. In response, the licensee stated no reduced transient fire HRRs were used in the MNGP fire PRA and therefore, the NRC staff finds that the licensee's use of heat release rates for transient fires is acceptable for this application.

The NRC requested information about whether obstructed plume modeling was used and if it was, then an indication of whether the base of the fire was assumed to be located at an elevation of less than one-half of the cabinet consistent with the guidance in NUREG-2178, Volume 1 "Refining and Characterizing Heat Release Rates from Electrical Enclosures During

Fire (RACHELLE–FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume" (ADAMS Accession No. ML15111A045). In response, the licensee stated that obstructed plume modeling in the MNGP fire PRA model used the guidance in NUREG/CR-6850 and therefore, the guidance of NUREG-2178 did not apply. The NRC staff determined that this response was adequate for this application.

The NRC requested the licensee identify what systems or components are assumed to always fail or not included in the fire PRA model and how these exclusions impact any of the RICT calculations. In response, the licensee identified nine SSC/systems that were excluded from the fire PRA model. The licensee determined that two of the SSC/systems (L-41 AC panel and standby liquid control system (SBLC)) impact RICT calculations. The response included incorporating these components in the fire PRA model used for RICT calculations as implementation items (Attachment 1 of the December 21, 2020, supplement to the LAR, implementation items 4 and 5, respectively). For the remaining seven items the licensee performed a sensitivity study that demonstrated that these items, in total, had negligible impact on RICT calculations. The NRC staff determined that the response was adequate for this application because the components that have the potential to impact RICT will be incorporated in the fire PRA model and the remaining components were demonstrated to not have an impact on this application.

The NRC staff requested information about how well-sealed cabinets were treated in the fire PRA. Information was requested about how fire propagation outside of well-sealed motor control centers (MCC) cabinets greater than 440 V was evaluated and whether it was consistent with the NRC guidance in Fire PRA FAQ 14-0009, "Treatment of Well-Sealed MCC Electrical Panels Greater than 440V" (ADAMS Accession No. ML15119A176). The NRC staff also requested explanation of whether well-sealed cabinets less than 440 V are included in the Bin 15 count of ignition sources consistent with guidance in NUREG/CR-6850. In response, the licensee stated that the MCCs are modeled in the MNGP fire PRA in accordance with NUREG/CR-6850. The NRC finds that the licensee's modeling of MCC cabinets is acceptable for this application.

The NRC requested information about whether incipient detection is credited in the fire PRA and whether its treatment is consistent with the most current NRC guidance. In response, the licensee stated that the MNGP does have incipient fire detection systems.

The NRC requested information about whether minimum joint human error probability (HEP) values less than 1E-05 were assumed in the fire PRA. NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report" (ADAMS Accession No. ML12216A104), discusses the need to consider a minimum value for the joint probability of human failure events (HFEs). NUREG-1921 refers to Table 2-1 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," (ADAMS Accession No. ML051160213) which recommends that joint human error probability (JHEP) values should not be below 1E-05. Table 4-4 of Electrical Power Research Institute (EPRI) 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a lower limiting value of 1E-06 for sequences with a very low level of dependence. Therefore, the guidance in NUREG-1921 allows for assigning JHEPs that are less than 1E-05, but only through assigning proper levels of dependency.

In response, the licensee stated that approximately 68 percent of the MNGP internal events, internal flooding, and fire JHEPs were below the 1E-05 guideline; however, the licensee did not detail how many fire PRA JHEPs were below the guideline and not provide examples with justification of the lower value. Instead, the licensee stated that the JHEPs below 1E-05 guideline will be reviewed and provide documented justification for the assigned values. The December 21, 2020, letter included an implementation item (item 7) to validate the JHEP values prior to implementing the RICT program. The NRC staff determined that the treatment of JHEP is appropriate for this application because values below the guidance threshold will be validated prior to implementing the RICT program.

In response to an RAI regarding fire modeling approaches, methods, and data, the licensee stated that the MNGP fire model uses NUREG-1805 fire dynamic tools, consolidated model of fire growth and smoke transport (CFAST), and the fire dynamic simulator (FDS) tools to assess fire modeling impacts. The NRC staff determined that the tools used in the MNGP fire modeling are consistent with state-of-practice as well as prior NRC approvals and are therefore appropriate for this application.

Regarding damage thresholds in support of the fire PRA analysis, the licensee stated that the MNGP fire analysis followed the approach detailed in Section H.2 of NUREG/CR-6850 and assumed all cabling in the plant to be thermoplastic. The NRC staff determined that the approach is an approved method and that the approach is appropriate for this application.

Section 5 of Enclosure 2 of the LAR stated that enhanced fire modeling methods, including heat soak, were used in the MNGP fire PRA model development. The December 21, 2020, supplement to the LAR confirmed that the heat soak method used is the one described in Appendix A of NUREG-2178, Volume 2. The NRC staff determined that the approach used is an approved method and appropriate for this application.

Based on its review, the NRC staff finds that the licensee's fire PRA has been peer reviewed against the current versions of the PRA standard and RG 1.200. Furthermore, the licensee has appropriately closed the F&Os or demonstrated that the open F&Os do not impact the application. The licensee will update the fire PRA model (implementation items 4, 5, and 7) prior to implementation of the RICT program. Therefore, after completion of the implementation items, the fire PRA is technically acceptable to support the RICT Program.

PRA Technical Adequacy Conclusion

Based on the NRC staff's review of the licensee's submittal and assessments, the NRC staff concludes that the MNGP PRA models for internal events, including internal flooding and for fire events used to implement the RICT program satisfy the guidance of RG 1.200, Revision 2. The NRC staff based this conclusion on the findings that the PRA models conform sufficiently to the applicable industry PRA standards for internal events including internal flooding and for fire events at an appropriate capability category, considering the licensee's acceptable disposition of the peer review of F&Os, the proposed implementation items, and NRC staff review.

Based on the review of the provided information, including completion of the implementation items, the MNGP PRA models are determined to be of sufficient technical adequacy to support implementation of the RICT program. Therefore, the NRC staff finds that the licensee has satisfied the intent of RG 1.177, Revision 1 (Sections 2.3.1, 2.3.2, and 2.3.3), and RG 1.174, Revision 3 (Sections 2.3 and 2.5), and that the MNGP PRA acceptability including completion of the implementation items will be sufficient to implement RMTS in accordance with NEI 06-09-A.

PRA Update Process

Section 4.0 of the SE for the NEI 06-09 states that an LAR should provide a discussion of the licensee's programs and procedures to ensure that the PRA models that provide the foundation for the Real Time Risk (RTR) model are maintained consistent with the as-built, as-operated plant. In the LAR, the term RTR model and CRMP model are both used and refer to the same model. Enclosure 7 of the LAR described a periodic update and review process for the PRAs that are used in the RTR model. The NRC staff reviewed the MNGP PRA model update process to assess if the PRA models that support the RICT program are maintained consistent with the as-built, as-operated and maintained plant.

The LAR indicated that the update process is consistent with NEI 06-09-A. Section 4.2 of LAR Enclosure 7 "PRA Model Update Process" explains that the MNGP PRA update requirements include: (1) review of plant changes and discovered conditions for potential impact on the PRA models and the RTR model including risk calculation to support the RICT program (e.g., plant changes, plant or industry operational experience, and errors or limitation identified in the modeling), (2) review of plant changes that meet the plant procedure criteria for updating the PRA models before the periodic update, (3) periodic update of the PRA models at least every two recycling outages, and (4) performance of interim risk analyses or implementation of administrative restrictions on use of the RICT program if significant plant changes or discovered conditions cannot be addressed immediately.

Section 2.3.4 of NEI 06-09-A specifies that "criteria shall exist in PRA configuration risk management to require PRA model updates concurrent with implementation of facility changes that significantly impact RICT calculations." The NRC noted that according to Enclosure 7 of the LAR, if plant changes or discovered conditions meet criteria defined in the plant PRA update procedures, then an unscheduled PRA update will be performed to incorporate the change. The NRC staff requested explanation of the conditions that must exist and criteria that would be used to require an unscheduled PRA update. In response, the licensee stated that it uses a living model to determine the impact of model changes that are not incorporated in the application model of record. The licensee further stated that the living model is quantified and assessed on a quarterly basis. It is assessed for a predictive significant impact on the RICT application. The response defines a significant impact as greater than a 25 percent change in overall plant risk (CDF or LERF). The NRC staff requested clarification of the quantitative and qualitative criteria that would supplement this criterion. In response, the licensee stated that quantification includes evaluation of the impact on the internal event (including flooding) and fire models, as well as all hazards. If any of these are affected by more than 25 percent, then entrance into RICTs is suspended unless the impacted RICTs remain conservative. Qualitative assessment is also applied when a modification, enhancement, or PRA model error is identified using the experience and judgment of the PRA analyst. The quarterly quantification includes a qualitative review of open PRA model change items. The NRC staff determined that this level of review is appropriate for this application and concludes the licensee's PRA model update process is consistent with RG 1.200, consistent with the guidance in NEI 06-09-A and, therefore, acceptable.

Risk Assessment Approaches and Methods

Changes to the PRA are expected to occur over time to reflect changes in PRA methods and changes to the as-built, as-operated, and maintained plant to reflect the operating experience at the plant as specified in RG 1.200, Revision 2. Changes in PRA methods are addressed by constraints of TS Administrative Section 5.5.16:

> The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

The NRC staff finds that this constraint is acceptable because it adequately implements the RICT program using models, methods, and approaches, consistent with applicable guidance that are acceptable to the NRC.

PRA Acceptability Conclusion

The licensee (1) reviewed the PRA using endorsed guidance and adequately resolved all identified issues, (2) established a periodic update and review process to update the PRA and associated CRMP model to incorporate changes made to the plant and PRA methods and data consistent with the RICT program, and (3) will calculate RICTs using NRC-accepted PRA methods. Therefore, the NRC staff concludes that the licensee has and will maintain a PRA that is technically adequate to support implementation of the RICT program.

<u>Scope of the PRA</u>

NEI 06-09-A requires a quantitative assessment of the potential impact on risk due to impacts from internal and external events, including internal fires, internal floods, and other significant external events. As discussed previously in this section, the MNGP PRA used for the RICT program includes contributions from internal and external events, including internal flooding and fire events. In addition, the licensee provided a bounding estimate of the seismic CDF (SCDF) and seismic LERF (SLERF) and included those SCDF and SLERF values into the change-in-risk used to calculate RICTs consistent with the guidance in NEI 06-09-A. For external hazards for which a PRA is not available, the guidance in NEI 06-09-A allows for the use of bounding analysis of the risk contribution of the hazard for incorporation into the RICT calculation or justification for why the hazard is not significant to the RICT calculation.

NEI 06-09-A requires a quantitative assessment of the potential impact on risk due to impacts from internal and external events, including internal fires. As clarified in the SE on NEI 06-09-A, other sources of risk (i.e., seismic and other external events) must be quantitatively assessed if they contribute significantly to the incremental risk of any RMTS configuration. Sources of risk shown to be insignificant contributors to configuration risk may be excluded for the RICT calculations. Additionally, shutdown risk assessment is not applicable to this LAR since the LAR only applies to Modes 1 and 2.

LAR Enclosure 4, “Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models,” provided the assessment of external hazard risk for the RICT program. LAR Enclosure 4 states that this assessment is based on an update of the MNGP individual plant examination of external events (IPEEE) external hazard screening evaluation. The LAR states that the hazards assessed in LAR Enclosure 4, Table E4-2, are those identified for consideration in non-mandatory Appendix 6-A of the PRA standard, which provides a guide for identification of most of the possible external events for a plant site. The NRC staff notes that this list is essentially the same list of hazards as presented in Table 4-1 of NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision-making,” dated March 2017 (ADAMS Accession No. ML17062A466). According to the LAR, the following external hazards were evaluated:

- Aircraft Impact
- Avalanche
- Biological Event
- Coastal Erosion
- Drought
- External Flooding and Intense Precipitation (further discussed in the LAR)
- Extreme Wind or Tornados (further discussed in the LAR)
- Fog
- Forest or Range Fire
- Frost
- Hail
- High Summer Temperature
- High Tide, Lake Level, or River Stage
- Hurricane
- Ice Cover
- Industrial or Military Facility Accident
- Internal Fire (evaluated in an internal fire PRA)
- Internal Flooding (evaluated in the internal events PRA)
- Landslide
- Lightning
- Low Lake Level or River Stage
- Low Winter Temperature
- Meteorite/Satellite Strike
- Pipeline Accident
- Release of Chemicals from On-site Storage
- River Diversion
- Sand or Dust Storm
- Seiche
- Seismic Activity (treated by adding the bounding seismic risk to the RICT calculations)
- Snow
- Soil Shrink-Swell
- Storm Surge
- Toxic Gas
- Transportation Accidents
- Tsunami
- Turbine-Generated Missiles
- Volcanic Activity
- Waves

LAR Enclosure 4, Section 2, stated for the overall process, consistent with NUREG-1855, that external hazards may be addressed by (1) screening the hazard on low frequency of occurrence, (2) bounding the potential impact and including it in the decision-making, and (3) developing a PRA model to be used in the RMAT/RICT calculation. The LAR states that as part of this process the following two aspects of the external hazard contribution to risk should be considered.

- The first is the contribution from the occurrence of beyond design basis conditions, for example, winds greater than design, seismic events greater than 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 is protected, the occurrence of these conditions nevertheless causes a demand on these systems that presents a risk.

LAR Table E4-2 provided a disposition for each non-seismic external hazard as well as other hazards and concludes that no unique PRA model for these hazards is required in order to assess configuration risk for the RICT program (with the exception of internal flooding and internal fire, which are addressed by a PRA).

The NRC staff notes that the initial preliminary screening criteria and progressive screening criteria presented in LAR Table E4-3 is the same criteria presented in supporting requirements EXT-B1 and EXT-C1 of the ASME/ANS PRA Standard for screening external hazards.

External Hazards

The NRC staff's SE in NEI 06-09-A states that sources of risk besides internal events and internal fires (i.e., seismic and other external events) must be quantitatively assessed if they contribute significantly to configuration-specific risk. The SE further states that bounding analyses or other conservative quantitative evaluations are permitted where realistic PRA models are unavailable. In addition, the SE concludes that if sources of risk can be shown to be insignificant contributors to configuration risk, then they may be excluded from the RMTS.

Enclosure 4 to the LAR addressed the risk from seismic events and other external hazards in the context of this application. The enclosure provides the basis for exclusion of certain hazards from consideration in the determination of RICTs due to their insignificance to the calculation of configuration risk as discussed above. This enclosure also provided the bounding estimate for the risk from seismic events for use in determining the configuration risk for the RICTs identified in the LAR as discussed below. The NRC staff reviewed Enclosure 4 to the LAR and supplemental information to determine the acceptability of the consideration of risk from seismic events and other external hazards for this application.

Seismic Hazard

In its December 21, 2020 response to RAI 19, the licensee provided an updated risk contribution from seismic events using a “seismic penalty” approach. The licensee’s approach for including the seismic risk contribution in the RICT calculation is to add a fixed seismic CDF (SCDF) and seismic LERF (SLERF) value to each calculation for the proposed MNGP RICTs regardless of the plant configuration. To estimate a RICT the licensee proposed to add a SCDF contribution of 6.42E-06 per year and a SLERF contribution of 2.35E-06 per year to the configuration-specific delta risk contribution from internal events (including internal flooding) and internal fire events.

The proposed bounding SCDF estimate is based on the approach provided in NRC Generic Issue 199 (GI-199), “Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants, Safety/Risk Assessment,” dated August 2, 2010 (ADAMS Package Accession No. ML100270582). The analysis used a plant-level high confidence of low probability of failure (HCLPF) capacity of 0.19g referenced to peak ground acceleration (PGA). This value is based on the analysis performed in response to the Near-Term Task Force (NTTF) recommendation 2.1 expedited seismic evaluation process (ESEP), “Monticello Nuclear Generating Plant: Expedited Seismic Evaluation Process (ESEP) – Augmented Approach to Post-Fukushima Near-Term Task Force (NTTF) 2.1,” dated December 23, 2014 (ADAMS Accession No. ML14357A280). HCLPF is the capacity representing 95 percent confidence that the conditional probability of failure of an SSC is 5 percent or less. The uncertainty parameter for seismic capacity was represented by a combined beta factor of 0.4. For the seismic hazard curve, the approach used the MNGP review level ground motion (RLGM) developed to support the ESEP analysis. However, the SSCs in the emergency filter treatment building were determined to have an HCLPF of 0.134g (gravity), significantly lower than the plant HCLPF value of 0.19g. For the seismic hazard curve, the approach used the plant-specific seismic hazard curves developed in response to the Near-Term Task Force (NTTF) recommendation 2.1 (ADAMS package Accession No. ML14136A285). The NRC staff’s previous assessments dated July 8, 2015 (ADAMS Accession No. ML15175A336) of the licensee’s re-evaluated seismic hazard states that the licensee’s methodology was acceptable and that the re-evaluated hazard adequately characterized the site. The previous NRC staff conclusion on the re-evaluated hazard is applicable here because the same seismic hazard was used for this application. The NRC staff’s review finds that the method to determine the baseline SCDF is acceptable because it is consistent with the approach used in GI-199 and uses the latest seismic hazard and fragility information for the MNGP site.

Concerning the proposed bounding SLERF estimate, the licensee states in the December 21, 2020, supplement that the bounding SLERF estimate is based on the containment building HCLPF of 0.3g determined during the IPEEE, dated November 17, 1995, “Monticello Individual Plant Examination of External Events (IPEEE),” Revision 1 (ADAMS Accession No. ML20094P954), the updated site-specific hazard estimates, dated May 14, 2014, “MNGP 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 Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident” (ADAMS Accession No. ML14136A288), and the non-seismic conditional large early release probability. NUREG-1407, “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities” states that “[g]enerally, containment penetrations are seismically rugged; a rigorous fragility analysis is needed only at review levels greater than 0.3g, but a walkdown to evaluate for unusual conditions (e.g., spatial interactions, unique penetration configurations) is recommended." MNGP states in their IPEEE that "[f]or the seismic IPEEE, screening of the capacity of SSCs was performed at 0.3g" and "that even if a conservative assumption were made that all SSCs not screened out at 0.3g were unavailable, sufficient equipment would remain available to accomplish core cooling and containment functions." The NRC staff finds that the approach to determine the SLERF is adequate for this application.

The NRC staff finds that during RICTs for SSCs credited in the design basis to mitigate seismic events, the licensee's proposed methodology captures the risk associated with seismically induced failures of redundant SSCs because such SSCs are assumed to be fully correlated. By assuming full correlation, the seismic risk for those RICTs will not increase if one of the redundant SSCs is unavailable because simultaneous failure of all redundant trains would be assumed in a seismic PRA. During RICTs for SSCs not credited in the design-basis seismic event, but which could be used when credited SSCs fail, the proposed methodology for considering seismic risk contributions may be non-conservative because the seismically-induced failure of such SSCs during the RICT may not be included in the risk increase. However, the occurrence and degree of non-conservatism depends on the plant high confidence in low probability of failure (HCLPF) value used for the RICT calculations, as compared to the HCLPF values for such SSCs. The degree of non-conservatism will be low or nonexistent if the plant HCLPF value is lower than most or all SSCs impacted by a seismic event. During RICTs for SSCs that are not used to mitigate a seismic event, the proposed methodology for considering seismic risk contributions is conservative because the seismically induced failure of such SSCs would not result in a risk increase associated with the plant configuration during the RICT, but the baseline seismic risk is still included in the calculation.

LAR Section 4.2 of Enclosure 4 included the calculated total (i.e., across the entire hazard curve) seismically induced (therefore, unrecoverable) loss of offsite power (LOOP) frequency of 1.84E-05 per year for the MNGP, which is about 8 percent of the total unrecovered LOOP frequency addressed in the internal events PRA for the MNGP. The NRC staff evaluated the analysis and finds that the analysis adequately addresses the impact of seismically induced LOOP and finds that seismically induced LOOP has an insignificant impact on the RICT program calculations.

In summary, the NRC staff's review finds the proposal to use the SCDF contributions of 6.42E-06 per year and a SLERF contribution of 2.35E-06 per year acceptable as an addition to the configuration-specific delta CDF and delta LERF for the licensee's RICT program for the MNGP because: (1) the licensee used the most current site-specific seismic hazard information for the MNGP, (2) the licensee used an acceptably low plant HCLPF value of 0.19g consistent with the information for the MNGP in the ESEP evaluation, (3) the licensee determined a SLERF penalty based on convolving the MNGP plant-level HCLPF seismic capacity (0.19g), composite variability (βc of 0.4), and the plant limiting HCLPF for containment integrity (0.3g), with the new site-specific hazard estimates for plants in the CEUS and spectral ratios developed from the MNGP Seismic Hazard and Screening Report, and (4) the licensee adding the baseline seismic risk to RICT calculations, which assumes the fully correlated failures, is conservative for SSCs credited in seismic events, while any potential non-conservative results for SSCs that are not credited in seismic events is small or nonexistent, as discussed above.

Extreme Wind or Tornado Hazards

LAR Enclosure 4, Section 4, discusses the evaluation of the extreme winds and tornadoes impact on this application. The basis for the insignificant impact of extreme winds and tornados (including tornado-generated missiles) for this application relies on the design of SSCs and a tornado missile analysis. Table E4-2 of the same enclosure presents the screening criteria used to disposition the risk for the extreme wind and tornado hazards. Table E4-2 indicates that criterion "C1" (Event damage potential is < events for which plant is designed) and criterion "PS4" (the bounding mean CDF is < 1 E-06 per year) was used to screen the extreme wind and tornado hazard.

The LAR stated that wind damage is bounded by tornadoes and that the tornado wind speed corresponding to a 1E-06 per year exceedance frequency is less than the MNGP design wind speed value. The tornado missile hazard was ultimately screened out based on the structures that protect safe shutdown equipment, which are designed for two types of missiles: (1) large utility poles with a velocity of 200 miles per hour (mph), (2) a one ton missile, such as a motor vehicle at 100 mph, and the CDF associated with tornado missiles is determined to be less than 1.1E-07 per year.

In summary, the NRC staff's evaluation of the considerations of extreme winds and tornadoes for the MNGP finds that the extreme winds and tornado hazard has an insignificant contribution to configuration risk and can be excluded from the calculation of the proposed RICTs.

External Flooding and Intense Precipitation

LAR Enclosure Table E4-2 presented the screening criteria used to disposition the risk for the external flooding hazard. Table E4-2 indicates that criterion "C1" (Event damage potential is < events for which plant is designed) was used to screen external flooding and intense precipitation hazards.

The table entry states that the results of the MNGP flood hazard reevaluation report (FHRR), "Monticello Nuclear Generating Plant: Response to Post-Fukushima Near-Term Task Force (NTTF) Recommendation 2.1. Flooding – Flood Hazard Reevaluation Report," dated May 12, 2016 (ADAMS Accession No. ML16145A179) show that flooding from rivers and streams is bounded by the current licensing basis. Local intense precipitation (LIP) was evaluated in a focused evaluation to determine if the plant's current design basis bounds the reevaluated flood parameters. The NRC staff concluded in its April 12, 2018, letter, "Monticello Nuclear Generating Plant- Staff Assessment of Flooding Focused Evaluation" (ADAMS Accession No. ML18081A948), that MNGP has effective flood protection for LIP events.

In summary, the NRC staff's evaluation of the considerations of external flooding hazards for MNGP finds that the external flooding hazard has an insignificant contribution to configuration risk and can be excluded from the calculation of the proposed RICTs.

Other External Hazards

Besides the external flooding, intense precipitation, and high winds and tornados discussed above, the rationale for the insignificant impact of non-seismic external hazards and other hazards for the MNGP site was presented in Table E4-2 of Enclosure 4 to the LAR. Regarding the external hazard of snow, the LAR analysis stated that the highest recorded snowfall in the area was 46.5 inches with an estimated weight of 46.5 pounds per square foot (psf) with the design weight at 50 psf. The response to RAI 20 addressed the small margin between the design and snowfall record in that the record is for a town in the northern part of the state and the snowfall record of 16.2 inches in Minneapolis is the appropriate record for the analysis. The NRC staff determined the use of the Minneapolis snowfall record is appropriate for this application. The NRC staff's review of the information in the submittal finds that the contributions from the other external hazards have an insignificant contribution to configuration risk and can be excluded from the calculation of the proposed RICTs because they either do not challenge the plant or they are bounded by the external hazards analyzed for the plant.

External Hazards Conclusion

The NRC staff concludes that the approach for considering the impact of seismic events, non-seismic external hazards and other hazards for the MNGP in the RICT calculations is acceptable because the approach included a technically acceptable quantitative assessment of the seismic risk, configuration specific tornado missile risk consistent with the guidance in NEI 06-09-A, and demonstrated the insignificant contribution to configuration risk from other external hazards on the proposed RICTs.

Shutdown Risk

Shutdown risk is not applicable to this LAR since the LAR only applies to Modes 1 and 2.

PRA Scope Conclusions

According to the LAR, the proposed RICT program is only applicable to operational Conditions (or Modes) 1 and 2; therefore, risk evaluations for Modes 3, 4, and 5 are not relevant to the proposed change.

Based on the above, the NRC staff finds that the licensee has satisfied the intent of RG 1.177, Revision 1 (Section 2.3.2), and RG 1.174, Revision 2 (Sections 2.3 and 2.5), and that the scope of the PRA model and the use of a conservative analysis for seismic events is appropriate for this application.

PRA Modeling

Section 3.2.2 of NEI 06-09-A specifies that to evaluate a RICT for a given Required Action, the specific systems or components involved should be directly modeled in the PRA or, if not directly modeled, the functions directly correlated to the specific systems or components are modeled in the PRA. TSTF-505, Revision 2, also states that Required Actions for systems that do not affect CDF or LERF or for which a RICT cannot be quantitatively determined are not in scope of the program. The LAR identified, for each of the TS LCO Required Actions for which the RICT program is proposed to apply, the following: (1) the SSCs are included within the scope of the PRA models, or surrogate SSCs are modeled that bound the functions of the TS SSCs; (2) the success criteria parameters used to determine PRA functional determination and, if different from the design-basis success criteria, then the plant-specific analyses that justify use of the PRA success criteria; and (3) a commitment to update the PRA models to incorporate modeling of the RPS instrumentation associated with TS LCO 3.3.1.1.A, mechanical vacuum pump system and isolation instrumentation associated with TS LCO 3.3.5.1 G, the ADS and instrumentation associated with TS LCO 3.3.7.2 A, and the SBLC system associated with TS LCO 3.1.7.B.

During the NRC staff review of Table E1-1 of the LAR it was noted that the design criteria for TS LCO 3.6.1.7.A (suppression chamber-to-drywell vacuum breakers) is six vacuum breakers whereas the PRA success criteria is one vacuum breaker. . The letter dated December 21, 2020, provided implementation item 6 to ensure the proper vacuum breaker PRA success criteria is determined and incorporated into the PRA model prior to implementation of the RICT program.

Regarding TS LCO 3.7.2.A (emergency service water) the licensee stated in the LAR that the ESW system was not modeled in the PRA because "hydraulic analysis has been performed to show that ESW is not required to prevent CDF and LERF." However, in response to an RAI, the licensee reported that the existing analysis for screening ECCS room and pump motor cooling was inadequate and further analysis is required to justify screening the equipment from the PRA model. The licensee added an implementation item (item 9) to ensure that prior to implementation of the RICT Program, ECCS room and pump motor cooling are modeled in the PRA with sufficient detail to accurately calculate a RICT. The NRC staff finds the response acceptable because the PRA model will consider the appropriate success criteria and ECCS room cooling.

System and Surrogate Modeling

Table E1-1 in Enclosure 1 to the LAR, as supplemented: (1) identifies each TS LCO condition in scope of the RICT program and the SSCs covered by the LCO, as applicable, (2) indicates whether the SSC is modeled in the PRA, and (3) for the cases in which the SSCs are not explicitly modeled, provides an explanation of how the PRA uses surrogate events that bound the function(s) of the TS LCO SSC(s).

RAI 06 requested explanation of how I&C systems are modeled in the PRA models and justification that they are modeled in sufficient detail to support the RICT program. The response to RAI 06 provided by letter dated December 21, 2020, stated that subcomponents, such as relays and sensors, are individually modeled in the PRA for each of the train or channels, the failure rates use generic industry data, and provided an updated Table E1-2 in the that accurately reflects the RICT estimates. The NRC staff determined that the response provided reasonable assurance that the I&C system modeling in the PRA is appropriate for this application.

For digital I&C systems, given the PRA modeling uncertainties associated with crediting digital I&C systems, NRC requested the results of a sensitivity study or identification of RMAs that will be applied to certain LCO Conditions during a RICT. In response, the licensee provided the results of a sensitivity study where several failure probabilities related to the digital feedwater control system were significantly increased. The results demonstrate that there was minimal impact on the majority of TS LCO RICTs; however, four RICTs changed by one day. Given the four specific TS LCOs were not identified in the analysis, the NRC staff evaluated the impact of one day for those RICTs in the updated Table E1-2 that were less than 30 days. The analysis found that the TS LCO SSCs most likely to mitigate a loss of feedwater have a change in RICT values less than seven percent. Based on this analysis and the overly conservative study, the NRC staff determined that this source of model uncertainty did not significantly impact this application.

PRA Modeling Conclusions

The NRC staff reviewed the information provided by the licensee and concluded that the PRA modeling used to support the RICT program can appropriately model alignments of components during periods when the RICT will be calculated. Therefore, the NRC staff finds that the licensee has satisfied the intent of RG 1.177, Revision 1 (Section 2.3.3), and RG 1.174, Revision 3 (Section 2.3), and that the PRA modeling is appropriate for this application.

Key Assumptions and Sensitivity and Uncertainty Analyses

Using PRAs to evaluate TS changes requires consideration of the assumptions made within the PRA that can have a significant influence on the ultimate acceptability of the proposed changes. Risk-informed analyses of TS changes can be affected by uncertainties regarding the assumptions made during the PRA model's development and application. In general, the risk resulting from TS CT changes is expected to be relatively insensitive to most uncertainties because the uncertainties tend to affect similarly both the base case and the case with the TS equipment unavailable. The licensee considered PRA modeling uncertainties and their potential impact on the RICT program and identified, as necessary, applicable RMAs to limit the impact of these uncertainties.

Enclosure 9 of the LAR discussed key assumptions and sources of uncertainty. The licensee evaluated the MNGP PRA model to identify the key assumptions and sources of uncertainty for this application consistent with the RG 1.200 definitions, using sensitivity and importance analyses to place bounds on uncertain processes, to identify alternate modeling strategies, and to provide information to users of the PRA. The enclosure stated that the internal events PRA uncertainty analysis was performed based on guidance in NUREG-1855. The LAR explained that plant-specific assumptions and sources of modeling uncertainty identified from the internal events PRA notebooks were considered, as well as generic sources of uncertainty from the EPRI-1016737, "Treatment of Parameter and Modeling uncertainty for Probabilistic Risk Assessments." Additionally, the LAR explained that uncertainty analysis included an evaluation of the Level 2 internal events PRA using the 32 Level 2 PRA topics outlined in EPRI-1026511, "Practical Guidance of the Use of Probabilistic Risk Assessment in Risk-informed Applications with a Focus on the Treatment of Uncertainty."

LAR Enclosure 9 stated that for both the internal events and fire PRAs that "no specific uncertainty issues have been identified that would impact the RICT application," and no candidate key assumption and sources of uncertainty were presented in the LAR. RAI 09 stated that it was not clear to NRC staff what specific process and criteria was used to screen uncertainties from an initial comprehensive list of assumptions and sources of PRA modeling uncertainty (including those associated with plant specific features, modeling choices, and generic industry concerns), in order to conclude that no uncertainty issues could impact the RICT calculations. NRC staff also stated that it was not clear whether certain key assumptions and sources of uncertainty were initially identified but found to be unimportant. Therefore, the RAI requested description of the specific process and criteria used to screen uncertainties from an initial comprehensive list of assumptions and sources of PRA modeling uncertainty and to discuss sensitivity studies that were performed and their results showing that the key assumptions or sources of uncertainty has no impact on RICT calculations. The response to RAI 09.a, provided by letter dated December 21, 2020, explained that the process and criteria used in the uncertainty analysis for this application follows the EPR-1016737 process by reviewing all relevant PRA notebooks for sources of uncertainty and are evaluated to determine if they significantly impact any RICT calculation. The response to RAI 09.b stated that sources of uncertainty were screened using the following criteria if a quantitative sensitivity study was not performed:

- negligible probability of occurrence of the failure scenarios of concern (e.g., several orders of magnitude below other scenarios with similar impacts),
- the impacted system(s) having very low importance in the PRA results, and
- surrogate events that demonstrate the impact of a potential uncertainty had negligible risk significance.

Regarding the process and procedures for evaluation of uncertainties (RAI 09.c) the response stated that MNGP has two procedures and change forms that ensure that qualified personnel review and disposition each change.

As noted in RAI 9.d, during the review of the licensee's PRA uncertainty notebooks provided during the audit, the NRC staff noted three PRA assumptions that may impact the application but did not appear to be examined or dispositioned for the application and therefore requested additional information as described below.

The NRC staff noted one sensitivity study performed by the licensee was related to operators venting containment below 50 psig which would disable the RCIC system even though the emergency operating procedures direct operators to vent below 56 psig. The licensee responded to RAI 9.d that the plant operating procedures include guidance to operators to ensure that depressurization actions do not impact the operability of injection systems if they are needed to support key safety functions. The procedures also include specific direction in recovering RCIC in the case of an overpressure event. The RAI response therefore justified the licensee's conclusion that this containment venting is not a source of uncertainty for RICT calculations.

RAI 9.d noted one assumption that only rapidly evolving overpressure events lead to a rupture of containment, and gradually evolving events, like the loss of containment heat removal, would create smaller leaks in containment (which will avoid containment rupture). In response, the licensee performed a sensitivity study that demonstrated a twelve percent increase in CDF and a twenty percent increase in LERF. In the April 20, 2021 letter, the licensee added an implementation item (item 8) to add an overpressure containment rupture probability to the PRA consistent with the Individual Plant Examination failure mechanisms for gradual overpressurization events.

RAI 9.d also noted an assumption that RCIC is credited after battery depletion. In response, the licensee provided the results of a sensitivity study where the operator action failure probability was increased by a factor of three. This study showed some impact on the RICTs for TS LCO 3.3.5.1.B (ATWS-RPT instrumentation) and 3.8.4.B (DC sources). In response to the RAI, the licensee justified the HEP assigned to the operator actions to operate the RCIC pump following battery depletion is based on industry-standard methods. The licensee explained that these operator actions are proceduralized, are incorporated into the plant's emergency operating procedures using pre-staged equipment and operators are trained in the performance of this action. The human reliability analysis considered the time available to perform the action, the stress levels involved, and the availability of written procedures. Therefore, the licensee concluded that this does not constitute a key source of uncertainty.

The NRC staff determined these responses adequately address these three identified sources of uncertainty and their impact on RICT calculations.

With regards to the fire PRA, the licensee states, in Section 3 of LAR Enclosure 9, that it used guidance from NUREG-1855 and guidance for fire PRA development including NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities” (ADAMS Accession No. ML15167A411) to address the fire PRA uncertainty analysis. The licensee explains that plant specific assumptions and sources of modeling uncertainty identified from the fire PRA were considered, as well as generic industry sources of uncertainty from the EPRI TR 1026511. Though not documented in the LAR, NRC staff review during the audit found that the licensee also identified plant specific sources of fire PRA uncertainty. The licensee explained that the fire PRA uncertainties were organized by the fire PRA topics presented in NUREG/CR-6850. The licensee cited definitions from NUREG-1855 for the terms “credible assumption” and “consensus model,” and explained that it has used consensus modeling approaches to develop the fire PRA and that besides NUREG/CR-6850 it used guidance from more recently issued NUREGs pertaining to fire PRA and fire PRA FAQs.

The NRC staff’s review indicates (1) the licensee performed an adequate assessment to identify the potential sources of uncertainty, (2) the identification of the key assumptions and sources of uncertainty was appropriate and consistent with the guidance in NUREG-1855 and associated EPRI TR-1016737 and EPRI TR-1026511, and (3) the licensee will implement appropriate changes to the PRA model to address uncertainty (implementation item 8). Therefore, the NRC staff finds that the licensee has satisfied the guidance in RG 1.177, Revision 1 (Sections 2.3.4 and 2.3.5), and RG 1.174, Revision 3 (Section 2.2.2), and that the identification of assumptions and treatment of model uncertainties for risk evaluation of extended CTs is appropriate for this application and consistent with the guidance identified in NEI 06-09-A.

PRA Results and Insights

The proposed change implements a process to determine TS RICTs rather than specific changes to individual TS CTs. NEI 06-09-A requires periodic assessment of the risk incurred due to operation beyond the “front stop” CTs due to implementation of a RICT program and comparison to the guidance of RG 1.174, Revision 3, for small increases in risk.

As with other unique risk-informed applications, supplemental risk acceptance guidelines that complement the RG 1.174 guidance are appropriate. NEI 06-09-A requires that configuration risk be assessed to determine the RICT and establishes the criteria for ICDP and ILERP on which to base the RICT. An ICDP of 1E-5 and an ILERP of 1E-6 are used as the risk measures for calculating individual RICTs. These limits are consistent with NUMARC 93-01, Revision 4 A, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants” (ADAMS Accession No. ML111116A198). The use of these limits in NEI 06-09-A aligns the TS CTs with the risk management guidance used to support plant programs for the Maintenance Rule, and the NRC staff accepted these supplemental risk acceptance guidelines for RMTS programs in its approval of NEI 06-09-A.

NEI 06-09-A, as modified by the limitations and conditions in the associated SE, requires that the cumulative impact of implementation of an RMTS be periodically assessed and shown to result in: (1) a total risk impact below 1E-5/year for changes to CDF, (2) a total risk impact below 1E-6/year for changes to LERF, and (3) the total CDF and total LERF must be reasonably shown to be less than 1E-4/year and 1E-5/year, respectively. The licensee indicated in

Enclosure 5 of the LAR that the estimated total CDF and LERF meet the 1E-4/year CDF and 1E-5/year LERF criteria of RG 1.174, consistent with the guidance in NEI 06-09-A and that these guidelines will be satisfied whenever a RICT is implemented.

LAR Enclosure 5 presented estimates of the total MNGP CDF and LERF, which are summarized, in Table 1 below based in part on conservatively determined seismic CDF and LERF "penalty" values. The total CDF (i.e., the sum of the internal events including internal flooding, fire and SCDF) is 9.4E-05 per year. The total LERF (i.e., the sum of the internal events including internal flooding, fire and SLERF) is 7.02E-06 per year.

Accordingly, the CDF and LERF values for MNGP meet the RG 1.174 risk acceptance guidelines.

Table 1 Total Risk for the MNGP (1/year)

	Internal Event and Fire Risk	Seismic Risk[1]	Total Risk
CDF	6.4E-05	3.0E-05	9.4E-05
LERF	5.52E-06	1.5E-06	7.02E-06

Note:

1. Based on bounding "penalty" value presented in LAR Enclosure 4, Section 3 for the seismic hazard

The licensee has incorporated NEI 06-09-A in the RICT program of TS 5.5.16 and, therefore, can calculate the RICT consistently with its criteria and assesses the RICT program to assure any risk increases are small per the guidance of RG 1.174, Revision 3, and intent of RG 1.177, Revision 1. Also, estimate of the current total CDF and LERF meets the intent of the RG 1.174, Revision 3, acceptance guidelines. Therefore, the NRC staff finds that the licensee's RICT program is consistent with NEI 06-09-A guidance and is acceptable.

During the NRC staff review it was noted that discrepancies exist in the reported RICT estimates provided in Table E1-2 of Enclosure 1 of the LAR and updated MNGP PRA documents. The response to RAI 15 provided by letter dated December 21, 2020, supplied an updated Table E1-2 that reflected the latest MNGP RICT calculated values.

It was noted that several Table E1-2 of Enclosure 1 of the LAR TS LCO RICTs were not provided and notated with 'Note 1' that stated several quantification results exceed the RICT program risk caps, but that some configurations could be allowable. Regarding TS 3.5.1.D, the RICT calculation was in error and the appropriate value of 15 days was provided in the Table E1-2 update. For TSs 3.5.1E, 3.8.4.B, 3.8.7.A, and 3.8.7.B, the licensee provided RICT values for each of the possible configurations that demonstrated RICT time could range between 2 days (no voluntary entry) to 30 days. The NRC staff determined that the approach provided in the table note is appropriate for this application.

Implementation of the RICT Program

Because NEI 06-09-A involves the real-time application of PRA results and insights by the licensee, the NRC staff reviewed the description of programs and procedures associated with implementation of the RICT program in Enclosure 10 of the LAR. The administrative controls on the PRA and on changes to the PRA should provide confidence that the PRA results are reasonable, and the administrative controls on the plant personal using the RICT should provide confidence that the RICT Program will be applied appropriately.

The means for demonstrating the technical acceptability of the PRA models include assessment against the PRA standards and RG 1.200, which includes guidance for performing peer reviews and focused-scope peer reviews. The technical adequacy of the PRA models is discussed in Enclosure 2, "Information Supporting Consistency with Regulatory Guide 1.200," and Enclosure 7, "PRA Model Update Process," of the LAR. Enclosure 8, "Attributes of the CRMP Model," summarizes the changes made to the baseline PRA model for use in the online model, changes made to the baseline PRA model for translation to the online model, and states that changes made to the online model configuration files are controlled and documented by plant procedures.

NEI 06-09-A specifies that the RMTS risk assessment process should be integrated into station-wide work control processes and defines the necessary attributes of the RMTS program structure. In the conduct of RMTS, procedural guidance is required for conducting and using the results of the risk assessment. These procedures should specify the station functional organizations and personnel, including operations, engineering, work management and PRA personnel, responsible for each step of the procedures. The procedures should also clearly specify the process for calculating the applicable RICT, implementing RMAs, and conducting, reviewing, and approving decisions to exceed the front-stop CT and remove equipment from service.

Enclosure 10, "Program Implementation," of the LAR described the implementing programs and procedures and the associated personnel training. The licensee explained that a RICT program description and implementing procedures will be developed. 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-A. The program will be integrated with the existing online work control process. Entry into the RICT program will require management approval prior to preplanned activities and as soon as practicable following emergent conditions. These and other attributes that will be addressed in the RICT program are identified in the LAR. The NRC staff found that the licensee will establish appropriate programmatic and procedural controls for its RICT program, consistent with the guidance of NEI 06-09-A Section 3.2.1.

NEI 06-09-A specifies that stations implementing an RMTS program shall provide training, in the programmatic requirements associated with the RMTS program and of the individual RICT evaluations, to personnel responsible for determining TS operability decisions or conducting RICT assessments. Training of plant personnel shall be provided for those organizations with functional responsibilities for performing or administering the CRMP (or RTR) commensurate with each position's responsibilities, in accordance with 10 CFR 50.120(b)(3) and other applicable regulations, within the RICT program, as described in NEI 06-09-A.

Enclosure 10 of the LAR described the program for providing training to its staff. The licensee identifies the attributes that the RICT program procedures will address, which are consistent with NEI 06-09-A. The LAR also identified the categories of plant personnel that will be trained and the different types of training that the different categories of plant personnel receive. This includes detailed or Level 1 training for individuals who will be directly involved in the implementation of the RICT program, Level 2 training for plant management positions with authority to approve entry into the RICT program and other management and personnel who closely support the RICT program, and Level 3 training for personnel that need basic knowledge of RICT Program requirements and procedures.

The NRC staff reviewed the description of the training program provided in the LAR and concluded that the program is consistent with the training requirements set forth in NEI 06-09-A Section 2.3.3. Therefore, the NRC staff finds that the licensee has proposed acceptable administrative controls on the PRA and on the personnel that will use the RICT program.

Section 2.0 of Enclosure 8 to the LAR, described the process for translating the baseline model into the CRMP. Models are adjusted so they can reflect actual plant configuration and optimized for quantification speed. Plant procedures specify that an acceptance test is performed after every CRMP model update. This test verifies proper translation of the baseline PRA models and acceptance of all changes made to the baseline PRA models into the CRMP model. This test also verifies correct mapping of plant components to the basic events in the CRMP model. In response to an RAI, the licensee further described the benchmarking activities the licensee performs to confirm consistency of the real-time risk model results to the results of the baseline PRA model.

The NRC staff concludes that the CRMP model used to calculate the RICTs is acceptable because the underlying PRA models will remain acceptable and the acceptance test will verify the CRMP model is consistent with the underlying baseline PRA.

NEI 06-09-A requires that stations implementing an RMTS program shall provide training in the programmatic requirements associated with the RMTS program and of the individual RICT evaluations, to personnel responsible for determining TS operability decisions or conducting RICT assessments. Training of plant personnel shall be provided for those organizations with functional responsibilities for performing or administering the CRMP commensurate with each position’s responsibilities, in accordance with 10 CFR 50.120(b)(3) and other applicable regulations, within the RICT program, as described in NEI 06-09-A.

Section 4.0 of Enclosure 8 to the LAR, described the program for providing training to its staff responsible for development and maintenance of the CRMP tool. The NRC staff reviewed the description of the training program provided in the LAR, and concluded that the program is consistent with the training requirements set forth in NEI 06-09-A. Therefore, the NRC staff finds that the licensee has proposed acceptable administrative controls on the PRA and on the personnel that will use the RICT program.

3.1.4.2 Tier 2: Avoidance of Risk Significant Plant Configurations

The second tier provides that a licensee should provide reasonable assurance that risk significant plant equipment outage configurations will not occur when specific plant equipment is taken out of service in accordance with the proposed TS change.

NEI 06-09-A does not permit voluntary entry into high-risk configurations, which would exceed instantaneous CDF and LERF limits of 1E-3/year and 1E-4/year, respectively. It further requires implementation of RMAs when the actual or anticipated risk accumulation during a RICT will exceed one-tenth of the ICDP or ILERP limit. Such RMAs may include rescheduling planned activities to lower risk periods or implementing risk-reduction measures. The limits established for entry into a RICT and for RMA implementation are consistent with the guidance of NUMARC 93-01, Revision 4A, endorsed by RG 1.160, Revision 3, as applicable to plant maintenance activities. The RICT program requirements and criteria are consistent with the principle of Tier 2 to avoid risk-significant configurations.

Consistent with NEI 06-09-A, Enclosure 12 of the LAR identifies three kinds of RMAs (i.e., actions to provide increased risk awareness and control, actions to reduce the duration of maintenance activities, and actions to minimize the magnitude of the risk increase). The LAR Enclosure 12 also provides examples of RMAs for an unavailable diesel generator, offsite circuit, offsite circuit and diesel generator, DC electrical power subsystem, and low pressure ECCS injection/spray subsystem. The LAR explained that determination of RMAs is performed using plant procedures and involves both qualitative and quantitative considerations for specific plant configuration and the consideration of the practical means available to manage risk.

Based on the incorporation of NEI 06-09-A in the TS as discussed in Attachment 1 of the LAR, and because the proposed changes are consistent with the guidance of RG 1.174, Revision 2, and RG 1.177, Revision 1, the NRC staff finds the licensee's Tier 2 program is acceptable and supports the proposed implementation of the RICT program.

3.1.4.3 Tier 3: Risk informed Configuration Risk Management

The third tier provides that a licensee should develop a program that ensures that the risk impact of out of service equipment is appropriately evaluated prior to performing any maintenance activity.

NEI 06-09-A addresses Tier 3 guidance by requiring assessment of the RICT to be based on the plant configuration of all SSCs that might impact the RICT, including safety-related and non-safety-related SSCs. If a risk-significant plant configuration exists, based on the expectation of exceeding a threshold of one-tenth of the risk on which the RICT is based, compensatory measures and RMAs are required to be implemented. Therefore, the NRC staff finds that the RICT program provides an acceptable methodology to assess and address risk significant configurations. The NRC staff also finds that proposed changes will require reassessment of any plant configuration changes to be completed in a timely manner based on the more restrictive limit of any applicable TS action requirement or a maximum of 12 hours after the configuration change occurs.

Based on the incorporation of NEI 06-09-A in the TS, as discussed in Attachment 1 of the LAR, and because the proposed changes are consistent with the Tier 3 guidance of RG 1.177, Revision 1, the NRC staff finds that the proposed changes are acceptable.

3.1.4.4 Key Principle 4: Conclusions

The licensee has demonstrated the technical adequacy and scope of its PRA models, and that the models can support implementation of the RICT program for determining CTs. Proper consideration of key assumptions and sources of uncertainty have been made. The risk metrics are consistent with the approved methodology of NEI 06-09-A and the RICT program is controlled administratively through plant procedures and training. The RICT program follows the NRC-approved methodology in NEI 06-09-A. The NRC staff concludes that the RICT program satisfies the fourth key safety principle of RG 1.177 and is, therefore, acceptable.

3.1.5 Key Principle 5: Performance Measurement Strategies – Implementation and Monitoring Program

RG 1.177, Revision 1, and RG 1.174, Revision 3, establish the need for an implementation and monitoring program to ensure that extensions to TS CTs do not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common-cause mechanisms. The purpose of the implementation and monitoring program is to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change. Revision 3 of RG 1.174 states that monitoring performed in conformance with the Maintenance Rule, 10 CFR 50.65, can be used when the monitoring performed is sufficient for the SSCs affected by the risk-informed application. According to LAR Enclosure 11, the SSCs in the scope of the RICT program are also in the scope of the Maintenance Rule. Monitoring programs will provide for evaluation and disposition of unavailability impacts which will may be incurred from implementation of the RICT program.

Section 3.3.3 of NEI 06-09-A instructs the licensee to track the risk associated with all entries beyond the "front stop" CT, and Section 2.3.1 provides a requirement for assessing cumulative risk, including a periodic evaluation of any increase in risk due to the use of the RMTS program to extend the CTs. According to LAR Enclosure 11, as revised in the supplement dated June 30, 2021, the licensee calculates cumulative risk every refueling cycle not to exceed 24 months, which is consistent with NEI 06-09-A. The licensee converts the cumulative ICDP and the ILERP into average annual values which are then compared to the limits of RG 1.174. If any limits are exceeded, corrective actions are taken to ensure that future plant operational risk is within the acceptance guidance. This evaluation assures that RMTS program implementation meets RG 1.174 guidance for small risk increases. The licensee is implementing NEI 06-09-A via the RICT program feature. The RICT program's risk-tracking feature is therefore acceptable and complies with this RMTS program.

The NRC staff concludes that the RICT program satisfies the fifth key safety principle of RG 1.177, Revision 1, and RG 1.174 by, in part, monitoring the average annual cumulative risk increase as described in NEI 06-09, Revision 0-A, and using this average annual increase to ensure the program as implemented meets RG 1.174 guidance for small risk increases and is therefore acceptable. Additionally, the NRC staff concludes that the RICT program satisfies the fifth key safety principle of RG 1.177, Revision 1, and RG 1.174 because, in part, all the affected SSCs are within the Maintenance Rule program which can be used to monitor changes to the reliability and availability of these SSCs.

3.2 Variations from TSTF-505

The NRC staff evaluated the proposed use of RICTs in the variations stated above in Section 2.2.4 in conjunction with evaluating the proposed use of RICTs in each of the individual LCO, Required Actions, and CTs stated above in Section 2.2.3. The NRC staff's evaluation of the licensee's proposed use of RICTs in the variations against the key safety principles is discussed above in Sections 3.1.1 through 3.1.5. Based on the above Sections 3.1.1 through 3.1.5, the NRC staff finds that each of the five key principles in RG 1.177, Revision 1, and RG 1.174, Revision 2, have been met and concludes that the proposed variations are acceptable.

3.2.1 Plant-Specific LCOs

The licensee identified in Section 2.4 of Attachment 1 in the LAR plant-specific LCOs for which NSPM is proposing to apply the RICT program that are variations from TSTF-505, Revision 2, for the following TSs:

- TS 3.5.1 – ECCS – Operating;
- TS 3.6.1.8 – RHR Drywell Spray; and
- TS 3.3.7.2 – Mechanical Vacuum Pump Isolation Instrumentation.

This section of the SE addresses the acceptability of variation options of TS LCO 3.5.1 and LCO 3.6.1.8. The evaluation of variation option of TS LCO 3.3.7.2 is addressed in Section 3.1.2.3 of this SE.

3.2.1.1 TS 3.5.1 – Emergency Core Cooling System (ECCS) – Operating

The ECCS is designed to limit the release of radioactive materials to the environment following a LOCA. The ECCS at the MNGP consists of the HPCI system, two CS pumps, four LPCI pumps, and three ADS valves. The TS LCO 3.5.1 requires that each ECCS injection/spray subsystem and the function of three ADS valves be OPERABLE.

The licensee indicated that TS LCO 3.5.1, Conditions B, C, D, and E, are plant-specific Conditions that are not in the NUREG-1433 STS and, therefore, not in TSTF-505, Revision 2. The following review is to address if the remaining OPERABLE ECCS subsystems could provide adequate core cooling during a design-basis LOCA for each of the TS LCO 3.5.1 Conditions B, C, D, and E, respectively. The review also addresses an error in LAR Table E1-1, "In-scope TS/LCO Conditions to Corresponding PRA Corresponding Functions," related to the operable ADS valves for LCO 3.5.1 Condition K.

Condition B: 1 LPCI subsystem (2 LPCI pumps) inoperable for reasons other than Condition A, or 1 CS pump inoperable

Condition B applies to either 1 LPCI subsystem (2 LPCI pumps) inoperable for reasons other than 1 LPCI pump inoperable, or one C pump inoperable. As indicated in LAR Table E1-1 and the information in the response to RAI 22 provided by letter dated December 21, 2020, the remaining OPERABLE ECCS subsystems in Condition B consist of either: (1) 2 LPCI pumps, 2 CS pumps, 3 ADS valves and HPCI, or (b) 4 LPCI pumps, 1 CS pump, 3 ADS valves and HPCI. Based on the review of the RAI 22 response, the NRC staff found that for Condition B, the applicable analysis of record (AOR) for the limiting design-basis LOCA case is a large break in the recirculation line with the single failure of a loss DC battery presented in USAR Table 14.7-11. The AOR of the limiting recirculation line break shows in the RAI 22 response that the remaining operable ECCS subsystems (1 CS pump, 2 LPCI pumps along with 3 ADS valves) are adequate to reflood the vessel and maintain core cooling and preclude fuel damage.

In reviewing the licensee's response, the NRC staff noted that the NRC previously reviewed and concluded that the AORs satisfied the ECCS performance criteria in 10 CFR 50.46 and were acceptable for use in supporting an extended power uprate (EPU) using GE methods for the General Electric fuel dated December 9, 2013 (ADAMS Accession No. ML13316B298) and using AREVA methods for the transitions to ATRIUM 10XM fuel dated June 5, 2015 (ADAMS Accession No. ML15072A141) and February 23, 2017 (ADAMS Accession No. ML16342B276).

The NRC staff also noted that the operable ECCS subsystems assumed in the AOR for an applicable limiting LOCA are exceeded by the ECCS capability retained by TS LCO 3.5.1 Conditions B; therefore, the NRC staff finds that the more restrictive AOR for the applicable limiting LOCA provides reasonable assurance that Condition B would not result in loss of core cooling function, and a RICT is appropriate.

Condition C: 1 LPCI pump in both LPCI subsystems inoperable

Condition C applies to 1 LPCI pump inoperable in each of the two LPCI subsystems. For the ECCS at the MNGP, each of the LPCI subsystems contains two pumps. As indicated in LAR Table E1-1 and the licensee's RAI 22 response, the operable ECCS subsystems in Condition C contain 2 LPCI pumps, 2 CS pumps, 3 ADS valves and HPCI. Based on the review of the licensee's RAI 22 response, the NRC staff noted that for Condition C, the applicable AOR for the limiting design basis LOCA case is a large break in the recirculation line with the single failure of a loss DC battery presented in USAR Table 14.7-11. The AOR of the limiting recirculation line break shows that the remaining ECCS subsystems (1 CS pump, 2 LPCI pumps along with 3 ADS valves) are adequate to reflood the vessel and maintain core cooling and preclude fuel damage. The NRC staff also noted that the operable ECCS subsystems assumed in the AOR for an applicable limiting LOCA are exceeded by the ECCS capability retained by TS LCO 3.5.1 Conditions C; therefore, the NRC staff finds that the more restrictive AOR for the applicable limiting LOCA provides reasonable assurance that Condition C would not result in loss of core cooling function and a RICT is appropriate.

Condition D: 2 LPCI subsystems (4 LPCI pumps) inoperable for reasons other than Condition C or G

Condition D applies to both LPCI subsystems inoperable for reasons other than Condition C or Condition G (due to open RHR intertie return isolation valve(s)). As indicated in LAR Table E1-1 and the licensee's RAI 22 response, the remaining operable ECCS subsystems in Condition D contain either 2 CS pumps, HPCI and 3 ADS valves Based on the review of the licensee's RAI 22 response, the NRC staff noted that for Condition D, the applicable AOR for the limiting design-basis LOCA case is a large break in the recirculation line with the single failure of LPCI injection valve failure presented in USAR Table 14.7-11. The AOR of the limiting recirculation line break shows that the remaining ECCS subsystems (2 CS pumps, HPCI along with 3 ADS valves) are adequate to reflood the vessel and maintain core cooling and preclude fuel damage. The NRC staff also noted that the operable ECCS subsystems assumed in the AOR for an applicable limiting LOCA are the same as the ECCS capability retained by TS LCO 3.5.1 Conditions D; therefore, the NRC staff finds that the AOR for the applicable limiting LOCA provides reasonable assurance that Condition D would not result in loss of core cooling function, and a RICT is appropriate.

Condition E: 1 CS subsystem (1 CS pump) inoperable and 1 LPCI subsystem (2 LPCI pumps) inoperable; or 1 CS subsystem (1 CS pump) inoperable and 1 or 2 LPCI pump(s) inoperable

Condition E applies to one CS subsystem inoperable and one LPCI subsystem inoperable, or, one CS subsystem inoperable and one or two LPCI pump(s) inoperable. As indicated in LAR Table E1-1 and the licensee's RAI 22 response, the remaining operable ECCS subsystems in Condition E contain either (1) 1 CS pump, 2 LPCI pumps, HPCI and 3 ADS valves, or (2) 1 CS pump, 2 or 3 LPCI pumps, HPCI and 3 ADS valves. Based on the review of the licensee's RAI 22 response, the NRC staff noted that for Condition E, the applicable AOR for the limiting design basis LOCA case is a large break in the recirculation line with the single failure of DC Battery presented in USAR Table 14.7-11. The AOR of the limiting recirculation line break shows that the remaining ECCS subsystems (1 CS pumps, 2 LPCI pumps and 3 ADS valves) are adequate to reflood the vessel and maintain core cooling and preclude fuel damage. The NRC staff also noted that the operable ECCS subsystems assumed in the AOR for an applicable limiting LOCA would not exceed the ECCS capability retained by TS LCO 3.5.1 Conditions E; therefore, the NRC staff finds that the AOR for the applicable limiting LOCA has provided reasonable assurance that Condition E would not result in loss of core cooling function, and a RICT is appropriate.

Condition K: 1 ADS valve inoperable

Table E1-1 of the LAR lists TS LCO 3.5.1.K as a condition with 1 ADS valve inoperable, and states that in the column of "Design Success Criteria" that along with other operable ECCS subsystems 3 ADS valves are available. The NRC staff found that the number of operable ADS valves in the statement is inconsistent with the ECCS at MNGP. Since a total of 3 ADS valves are considered available for mitigating the consequences of LOCAs, the number of operable ADS valves under Condition K with 1 ADS valve inoperable should be 2 instead of 3. In RAI 23, the NRC staff requested the licensee to clarify how many operable ADS valves were credited in the AOR for LOCA in support of TS 3.5.1.K Condition regarding the adequacy of maintaining core cooling function. In its response, the licensee clarified that the operable ECCS subsystems in Condition K consist of 2 (instead of 3) ADS valves, 2 CS pumps, HPCI, and 4 LPCI pumps. In addition, the licensee indicated that the operable ECCS subsystems in Condition K bounds that assumed in the AOR for an applicable limiting design-basis LOCA case, a large break in the recirculation line with the single failure of a loss of ADS valve (in USAR Table 14.7-11). The AOR of the applicable limiting LOCA showed that the remaining ECCS subsystems (2 ADS valves, 2 CS pump, HPCI, and 4 LPCI pumps) are adequate to reflood the vessel and maintain core cooling and preclude fuel damage. Since the operable ECCS subsystems assumed in the AOR for an applicable LOCA are equivalent to the ECCS capability retained by TS LCO 3.5.1 Condition K, the NRC staff concludes that the AOR for the applicable limiting LOCA provides reasonable assurance that Condition K would not result in loss of core cooling function, and a RICT is appropriate.

3.2.1.2 TS 3.6.1.8 – RHR Drywell Spray

The RHR drywell spray system is designed to condense any steam that may exist in the drywell; and thereby reduce drywell pressure and temperature following a DBA. At the MNGP, the RHR drywell spray mode of operation is not credited in the LOCA. However, it is credited for the evaluation of steam line breaks (SLBs) inside the drywell. The RHR drywell spray system contains two redundant RHR drywell spray subsystems. TS LCO 3.6.1.8 requires that two RHR drywell spray subsystems be operable, and the associated Condition A allows one RHR drywell spray subsystem inoperable.

The licensee indicated that MNGP TS 3.6.1.8 Condition A is a plant-specific condition not in the NUREG-1433 STS and, therefore, not in TSTF-505, Revision 2.

Condition A applies to one RHR drywell spray system inoperable. The MNGP safety analyses takes credit for the operation of the drywell spray function. As stated in the MNGP USAR, Section 5.2.3.9, "Drywell Temperature Analysis for Drywell Wall Temperature," a minimum of one RHR drywell spray subsystem is required to mitigate the consequences of SLBs in the drywell and maintain the primary containment peak temperature below the design limit.

Therefore, the NRC staff concludes that TS 3.6.1.8 Condition A is an adequate MNGP plant-specific LCO Condition since: (1) the analysis supporting the LCO conditions satisfies the GDC 50 requirements insofar as it relates to the containment pressure and temperature limits during DBAs, and (2) the LCO conditions satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii) insofar as it relates to the primary success path that functions or actuates to mitigate a DBA.

3.2.1.3 Plant Specific LCOs Conclusion

The NRC staff has reviewed the acceptability of variation options of the relevant LCOs and associated actions of TSs 3.5.1 and 3.6.1.8 and concluded that TS 3.5.1 Conditions B, C, D, E and K, and LCO 3.6.1.8 Condition A meet the requirements for inclusion in the RICT Program and are acceptable, since (1) the LCO conditions satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii) insofar as it relates to the primary success path that functions or actuates to mitigate a DBA, and (2) the LOCA analysis supporting the LCO conditions satisfies 10 CFR 50.46 insofar as it relates to the ECCS performance acceptance criteria.

The NRC staff evaluated the proposed use of RICTs in the variations stated above in Section 2.2.4 in conjunction with evaluating the proposed use of RICTs in each of the individual LCO actions and CTs stated above in Section 2.2.3. The NRC staff's evaluation of the licensee's proposed use of RICTs in the variations against the key safety principles is discussed above in Sections 3.1.1 through 3.1.5. Based on the above Sections 3.1.1 through 3.1.5, the NRC staff finds that each of the five key principles in RG 1.177, Revision 1, and RG 1.174, Revision 2, has been met and concludes that the proposed variations are acceptable.

3.2.2 Proposed Changes to TSs Not Associated with TSTF-505, Revision 2

Section 2.2.4.3 of this SE discusses changes proposed in the LAR that are not related to TSTF-505 adoption. The NRC staff finds these proposed changes are editorial because the changes: (1) do not involve any physical changes to the structures, systems, or components or the way that the unit is operated and controlled, (2) do not affect the technical content or operational requirements in the TSs, and (3) do not affect provisions relating to organization and management, procedures, recordkeeping, review and audit, nor reporting necessary to assure operation of the facility in a safe manner. Therefore, the NRC staff determined that the requirements of 10 CFR 50.36(c)(5) and 10 CFR 50.36(c)(2)(i) for these TS LCOs will continue to be met and that the remedial actions proposed in these TSs can be followed by the licensee until the LCO can be met or if the remedial actions cannot be met within the CTs the licensee will be required to shut down the reactor. Therefore, the NRC staff concludes that the proposed changes are acceptable.

3.3 TS Administrative Controls Section

The NRC staff reviewed the licensee's proposed addition of a new program, the RICT program, to the Administrative Controls section of the TS. The NRC staff evaluated the elements of the new program to ensure alignment with the requirements in 10 CFR 50.36(c)(5) and to ensure the programmatic controls are consistent with the RICT program described in NEI 06-09-A.

TS 5.5.16 requires that the RICT program be implemented in accordance with NEI 06-09-A. This is acceptable because NEI 06-09-A establishes an appropriate framework for an acceptable RICT program.

The TS states that a RICT may not exceed 30 days. The NRC staff determined that 30-day limit is appropriate because it allows sufficient time to restore SSCs to operable status while avoiding excessive out of service times for TS SSCs.

The TS states that the RICT may only be used in Modes 1 and 2. This provision ensures that the RICT is only used for determination of CDF and LERF for modes of operation modeled in the PRA.

The TS requires that while in a RICT, any change in plant configuration as defined in NEI 06-09-A must be considered for the effect on the RICT. The TS also specifies time limits for determining the effect on the RICT. These time limitations are consistent with those specified in NEI 06-09-A.

The TS contains requirements for the treatment of CCFs for emergent conditions in which the common cause evaluation is not complete. The requirements are to either: (1) numerically account for the increased probability of CCF or (2) to implement RMAs that support redundant or diverse SSCs that perform the functions of the inoperable SSCs and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs. Key Principle 2 of risk informed decision making is to assure that the change is consistent with DID philosophy. The seven considerations supporting the evaluation of the impact of the change on DID are discussed in RG 1.174, including one to preserve adequate defense against potential CCF. The NRC staff finds that numerically accounting for an increased probability of failure will shorten the estimated RICT based on the particular SSCs involved thereby limiting the time when a CCF could affect risk. Alternatively, implementing actions that can increase the availability of other mitigating SSCs or decrease the frequency of demand on the affected SSCs will decrease the likelihood that a CCF could affect risk. The NRC staff concludes that both the quantitative and the qualitative actions minimize the impact of CCF and therefore support meeting Key Principle 2 as described in RG 1.174. These methods either limit the exposure time, help ensure the availability of alternate SSCs, or decrease the probability of plant conditions requiring the safety function to be performed. The NRC staff finds that these methods contribute to maintaining DID because the methods limit the exposure time or ensure the availability of alternate SSCs.

The TS contains a provision that risk assessment approaches and methods used shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in RG 1.200, Revision 2. Methods to assess the risk from extending the CTs must be PRA methods used to support this LAR, or other methods approved by the NRC for generic use. As stated in the NRC staff's SE of NEI 06-09-A:

> TR NEI 06-09, Revision 0, requires an evaluation of the PRA model used to support the RMTS against the requirements of RG 1.200, Revision 1, and ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications", for capability Category II. This assures that the PRA model is technically adequate for use in the assessment of configuration risk. This capability category of PRA is sufficient to support the evaluation of risk associated with out of service SSCs and establishing risk informed CTs.

TS 5.5.16 was updated to reflect the current revision of RG 1.200. RG 1.200 incorporates ASME RA-S-2002 by reference.

The NRC staff's SE of NEI 06-09-A also states:

> As part of its review and approval of a licensee's application requesting to implement the RMTS, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods approved by the NRC staff for use in the plant-specific RMTS program. If a licensee wishes to change its methods, and the change is outside the bounds of the license condition, the licensee will need NRC approval, via a license amendment, of the implementation of the new method in its RMTS program. The focus of the NRC staff's review and approval will be on the technical adequacy of the methodology and analyses relied upon for the RMTS application.

This limitation and condition is being relocated from a license condition to the Administrative Controls section of the TS. Proposed TS 5.5.16 restates this limitation and condition from the NRC staff's SE in language that is appropriate for the Administrative Controls section of the MNGP TS. This constraint appropriately requires the licensee to utilize the risk assessment approaches and methods previously approved by the NRC and/or incorporated in the RICT program, and requires prior NRC approval for any change in PRA methods to assess risk that are outside those approval boundaries. The NRC staff finds that this requirement is appropriately reflected in the Administrative Controls section of the MNGP TS.

The regulations in 10 CFR 50.36(c)(5) require the TS to contain Administrative Controls providing "provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." The NRC staff has determined that the Administrative Controls section of the TS will assure operation of the facility in a safe manner when the facility uses the RICT program. Therefore, the NRC staff has determined that the requirements of 10 CFR 50.36(c)(5) are satisfied.

3.4 Technical Evaluation Conclusions

The NRC staff has evaluated the proposed changes against each of the five key principles in RG 1.177, Revision 1, and RG 1.174, Revision 2.

The proposed changes to the LCO conditions and the CTs for remedial actions are acceptable and will continue to meet 10 CFR 50.36(c)(2), 50.57(a)(2), and 50.57(a)(6). Therefore, the NRC staff concludes that the proposed change meets Key Principle 1: the change meets current regulations.

For LCO conditions in the existing TS, some reduction in defense in depth has already been evaluated and accepted for a limited period of time during the current CT, and the RICT Program provides solely a risk informed extension for operating in that plant condition. Therefore, the NRC staff concludes that the proposed change meets Key Principle 2: change is consistent with defense in depth philosophy.

Implementation of the methodology as described in TS 5.5.16 provides confidence that the CTs can be extended without any unanalyzed reductions in safety margins because the design basis success criteria parameters will be at the same level and provided by the same equipment as has been currently accepted. Therefore, the NRC staff concludes that the proposed change meets Key Principle 3: maintains sufficient safety margins.

The LAR has demonstrated the technical acceptability and scope of the PRA models and that the models can support implementation of the RICT Program for determining the identified CTs. The risk metrics will be consistent with the NRC approved methodology of NEI 06 09 A; RG 1.174, Revision 2; RG 1.177, Revision 1; and the RICT Program is controlled administratively through plant procedures and training. Therefore, the NRC staff concludes that the proposed change meets Key Principle 4: proposed increases in CDF or risk are small and are consistent with the Commission's Safety Goal Policy Statement.

The licensee's PRA model takes the sum of the contributors to risk associated with each application of the RICT Program, and that change in CDF or LERF above the zero maintenance baseline levels is converted into average annual values which are then compared to the limits of RG 1.174. If any limits are exceeded, corrective actions are taken to ensure future plant operational risk is within the acceptance guidance. The SSCs in the scope of the RICT Program that have their CTs extended by entry into the RICT Program are monitored to ensure their safety performance is not degraded because the SSCs in the scope of the RICT Program are also in the scope of the Maintenance Rule. Revision 2 of RG 1.174 states that monitoring performed in conformance with the Maintenance Rule, 10 CFR 50.65, can be used when the monitoring performed is sufficient for the SSCs affected by the risk informed application. The NRC staff, therefore, concludes that the proposed change meets Key Principle 5: use performance measurement strategies to monitor the change.

The NRC staff concludes that the proposed changes satisfy the key principles of risk informed decision making identified in RG 1.174, Revision 2, and RG 1.177, Revision 1, and, therefore, the requested adoption of the proposed changes to the TSs, implementation items, and associated guidance is acceptable.

4.0 ADDITIONAL CHANGES TO THE OPERATING LICENSE

The implementation of the amendment shall include the items listed in Table A5-1, "RICT Program PRA Implementation Items" of the NSPM letter dated April 20, 2021.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment on June 4, 2021. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding published in the *Federal Register* on May 19, 2020 (85 FR 29985). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

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Date of Issuance: July 12, 2021

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT NO. 206 RE: TSTF-505, REVISION 2, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES – RITSTF INITIATIVE 4b" (EPID L-2020-LLA-0062) DATED JULY 12, 2021

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