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

March 15, 2021

Mr. Christopher P. Domingos
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 235 AND 223 RE: ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE (TSTF) TRAVELER TSTF-505, REVISION 2, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES – RITSTF INITIATIVE 4b" (EPID L-2019-LLA-0283)

Dear Mr. Domingos:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 235 to Renewed Facility Operating License No. DPR-42 and Amendment No. 223 to Renewed Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated December 16, 2019, as supplemented by letter dated September 1, 2020.

The amendments modify TS requirements to permit the use of risk-informed completion times in accordance with Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b."

A copy of our related safety evaluation is also enclosed. The 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 Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 235 to DPR-42
2. Amendment No. 223 to DPR-60
3. Safety Evaluation

cc: Listserv

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 235
Renewed License No. DPR-42

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

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated December 16, 2019, as supplemented by letter dated September 1, 2020, 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-42 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 235, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2021.03.15 12:48:36 -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: March 15, 2021

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 223
Renewed License No. DPR-60

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

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated December 16, 2019, as supplemented by letter dated September 1, 2020, 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-60 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 223, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2021.03.15 12:49:13 -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: March 15, 2021

ATTACHMENT TO LICENSE AMENDMENT NOS. 235 AND 223

RENEWED FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Renewed Facility Operating License Nos. DPR-42 and DPR-60 with the attached revised pages. The changed areas are identified by a marginal line.

Renewed Facility Operating License No. DPR-42

REMOVE	INSERT
Page 3	Page 3

Renewed Facility Operating License No. DPR-60

REMOVE	INSERT
Page 3	Page 3

Technical Specifications

Replace the following pages of the 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.

REMOVE	INSERT	REMOVE	INSERT	REMOVE	INSERT
1.3-15	1.3-15	3.3.1-23	3.3.1-23	--------	3.4.9-3
--------	1.3-16	3.3.1-24	3.3.1-24	3.4.11-2	3.4.11-2
--------	1.3-17	--------	3.3.1-25	3.4.11-3	3.4.11-3
3.3.1-1	3.3.1-1	--------	3.3.1-26	3.4.11-4	3.4.11-4
3.3.1-2	3.3.1-2	--------	3.3.1-27	--------	3.4.11-5
3.3.1-3	3.3.1-3	3.3.2-1	3.3.2-1	3.5.2-1	3.5.2-1
3.3.1-6	3.3.1-6	3.3.2-2	3.3.2-2	3.5.2-2	3.5.2-2
3.3.1-7	3.3.1-7	3.3.2-3	3.3.2-3	3.5.2-3	3.5.2-3
3.3.1-8	3.3.1-8	3.3.2-4	3.3.2-4	3.5.2-4	3.5.2-4
3.3.1-9	3.3.1-9	3.3.2-5	3.3.2-5	3.6.2-5	3.6.2-5
3.3.1-10	3.3.1-10	3.3.2-6	3.3.2-6	3.6.3-2	3.6.3-2
3.3.1-11	3.3.1-11	3.3.2-7	3.3.2-7	3.6.3-4	3.6.3-4
3.3.1-12	3.3.1-12	3.3.2-8	3.3.2-8	3.6.5-1	3.6.5-1
3.3.1-13	3.3.1-13	3.3.2-9	3.3.2-9	3.6.5-2	3.6.5-2
3.3.1-14	3.3.1-14	3.3.2-10	3.3.2-10	3.7.2-1	3.7.2-1
3.3.1-15	3.3.1-15	3.3.2-11	3.3.2-11	3.7.2-2	3.7.2-2
3.3.1-16	3.3.1-16	3.3.2-12	3.3.2-12	--------	3.7.2-3
3.3.1-17	3.3.1-17	--------	3.3.2-13	3.7.4-1	3.7.4-1
3.3.1-18	3.3.1-18	3.3.4-3	3.3.4-3	3.7.5-2	3.7.5-2
3.3.1-19	3.3.1-19	3.3.4-4	3.3.4-4	3.7.7-1	3.7.7-1
3.3.1-20	3.3.1-20	3.3.4-5	3.3.4-5	3.7.7-2	3.7.7-2
3.3.1-21	3.3.1-21	--------	3.3.4-6	--------	3.7.7-3
3.3.1-22	3.3.1-22	3.4.9-2	3.4.9-2	3.7.8-1	3.7.8-1

REMOVE	INSERT	REMOVE	INSERT	REMOVE	INSERT
3.7.8-2	3.7.8-2	--------	3.8.1-10	5.0-22	5.0-22
3.7.8-3	3.7.8-3	3.8.4-1	3.8.4-1	5.0-23	5.0-23
3.8.1-1	3.8.1-1	3.8.4-2	3.8.4-2	5.0-24	5.0-24
3.8.1-2	3.8.1-2	3.8.4-3	3.8.4-3	5.0-25	5.0-25
3.8.1-3	3.8.1-3	--------	3.8.4-4	5.0-26	5.0-26
3.8.1-4	3.8.1-4	3.8.7-1	3.8.7-1	5.0-27	5.0-27
3.8.1-5	3.8.1-5	3.8.7-2	3.8.7-2	5.0-28	5.0-28
3.8.1-6	3.8.1-6	3.8.9-1	3.8.9-1	5.0-29	5.0-29
3.8.1-7	3.8.1-7	3.8.9-2	3.8.9-2	5.0-30	5.0-30
3.8.1-8	3.8.1-8	--------	3.8.9-3	5.0-31	5.0-31
3.8.1-9	3.8.1-9	5.0-21	5.0-21		

(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 and equipment calibration or associated with radioactive apparatus or components;

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

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purpose of volume reduction and decontamination.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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 1677 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 235, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NSPM shall fully 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 Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains

(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 and equipment calibration or associated with radioactive apparatus or components;

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

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purposes of volume reduction and decontamination.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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) <u>Maximum Power Level</u>

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1677 megawatts thermal.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 223, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) <u>Physical Protection</u>

NSPM shall fully 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 Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

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

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

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-8 (continued)

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

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

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

IMMEDIATE COMPLETION TIME

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

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s) or train(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1 Restore channel to OPERABLE status.	48 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel or train inoperable.	C.1 Restore channel or train to OPERABLE status. <u>OR</u> C.2.1 Initiate action to fully insert all rods. <u>AND</u> C.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.	48 hours 48 hours 49 hours
D. One Power Range Neutron Flux channel inoperable.	----------------NOTE----------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels. -- D.1.1 Place channel in trip. <u>AND</u>	 6 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	D.1.2 ------------NOTE------------ Only required to be performed when THERMAL POWER is > 85% RTP and the Power Range Neutron Flux input to QPTR is inoperable. -------------------------------- Perform SR 3.2.4.2.	 Once per 12 hours
E. One channel inoperable.	----------------NOTE----------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. -- E.1 Place channel in trip.	 6 hours OR In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. One channel inoperable.	-----------------NOTE---------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. --- K.1 Place channel in trip.	 6 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. One or both channel(s) inoperable on one bus.	-----------------NOTE---------------- One inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ---	
	L.1 Place channel(s) in trip.	6 hours <u>OR</u> ------NOTE------- Not applicable when more than one channel inoperable on one bus. --------------------- In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
M. One Reactor Coolant Pump Breaker Open channel inoperable.	M.1 Restore channel to OPERABLE status.	48 hours <u>OR</u> ------NOTE------- Not applicable when THERMAL POWER is below P-8 and above P-7. --------------------- In accordance with the Risk Informed Completion Time Program
N. Required Action and associated Completion Time of Condition K, L, or M not met.	N.1 Reduce THERMAL POWER to < P-7 and P-8.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
O. One Turbine Trip channel inoperable.	-----------------NOTE---------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channel(s). ---	
	O.1 Place channel in trip.	6 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
P. Required Action and associated Completion Time of Condition O not met.	P.1 Reduce THERMAL POWER to < P-9.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Q. One train inoperable.	-----------------NOTE---------------- One train may be bypassed for up to 8 hours for surveillance testing provided the other train is OPERABLE. ---	
	Q.1 Restore train to OPERABLE status.	6 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
R. One RTB train inoperable.	----------------NOTES--------------- 1. One train may be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. 2. One RTB may be bypassed for up to 4 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE. ---	
	R.1 Restore train to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program
S. One or more channels inoperable.	S.1 Verify interlock is in required state for existing unit conditions.	1 hour
T. One or more channels inoperable.	T.1 Verify interlock is in required state for existing unit conditions.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
U. Required Action and associated Completion Time of Condition T not met.	U.1 Be in MODE 2.	6 hours
V. One trip mechanism inoperable for one RTB.	V.1 Restore trip mechanism to OPERABLE status.	48 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
W. Required Action and associated Completion Time of Condition B, D, E, Q, R, S, or V not met.	W.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.2	---------------------------NOTES--------------------------- 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. -- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	----------------------------NOTES--------------------------- 1. Adjust NIS channel if absolute difference is ≥ 2%. 2. Not required to be performed until 72 hours after THERMAL POWER is ≥ 15% RTP. --- Compare results of the core power distribution measurements to NIS AFD.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.4	----------------------------NOTE--------------------------- This Surveillance must be performed on the reactor trip bypass breaker prior to placing the bypass breaker in service. --- Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.6	-----------------------------NOTE---------------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 75% RTP. --- Calibrate excore channels to agree with core power distribution measurements.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.7	-----------------------------NOTE---------------------------- Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. --- Perform COT.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	------------------------------NOTES------------------------------ 1. This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. 2. Not required to be performed for intermediate and source range instrumentation prior to reactor startup following shutdown ≤ 48 hours. -- Perform COT.	--------NOTE------- Only required when not performed within the Frequency specified in the Surveillance Frequency Control Program ----------------------- Prior to reactor startup <u>AND</u> Twelve hours after reducing power below P-10 for power and intermediate range instrumentation <u>AND</u> Four hours after reducing power below P-6 for source range instrumentation <u>AND</u> In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	------------------------------NOTE------------------------------ Verification of setpoint is not required. -- Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.10	------------------------------NOTE------------------------------ This Surveillance shall include verification that the time constants are adjusted to the prescribed values. -- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.11	------------------------------NOTE------------------------------ Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.12	-----------------------------NOTE---------------------------- This Surveillance shall include verification of Reactor Coolant System resistance temperature detector bypass loop flow rate. --- Perform CHANNEL CALIBRATION.	 In accordance with the Surveillance Frequency Control Program
SR 3.3.1.13	Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.14	-----------------------------NOTE---------------------------- Verification of setpoint is not required. --- Perform TADOT.	 In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.15	-----------------------------NOTE---------------------------- Verification of setpoint is not required. --- Perform TADOT.	Prior to exceeding the P-9 interlock whenever the unit has been in MODE 3, if not performed within the previous 31 days
SR 3.3.1.16	-----------------------------NOTE---------------------------- Neutron detectors are excluded from response time testing. --- Verify RTS RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program

Table 3.3.1-1 (page 1 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1, 2	2	B	SR 3.3.1.14	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					
a. High	1, 2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 110% RTP
b. Low	1(b), 2	4	D	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 40% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1, 2	4	D	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 6% RTP with time constant ≥ 2 sec
b. High Negative Rate	1, 2	4	D	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 8% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F, G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 40% RTP

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(b) Below the P-10 (Power Range Neutron Flux) interlocks.
(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

Table 3.3.1-1 (page 2 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Source Range Neutron Flux	2[(d)]	2	H, I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 1.0E6 cps
	3[(a)], 4[(a)], 5[(a)]	2	I, J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 1.0E6 cps
6. Overtemperature ΔT	1, 2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.16	Refer to Note 1 (Page 3.3.1-23)
7. Overpower ΔT	1, 2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.16	Refer to Note 2 (Page 3.3.1-24)
8. Pressurizer Pressure					
a. Low	1[(e)]	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 1845 psig
b. High	1, 2	3	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 2400 psig

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 3 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Pressurizer Water Level - High	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 90%
10. Reactor Coolant Flow- Low	1(f)	3 per loop	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 91%
11. Loss of Reactor Coolant Pump (RCP)					
a. RCP Breaker Open	1(f)	1 per RCP	M	SR 3.3.1.14	NA
b. Under-frequency 4 kV Buses 11 and 12 (21 and 22)	1(f)	2 per bus	L	SR 3.3.1.9 SR 3.3.1.10	≥ 58.2 Hz
12. Undervoltage on 4 kV Buses 11 and 12 (21 and 22)	1(e)	2 per bus	L	SR 3.3.1.9 SR 3.3.1.10	≥ 76% rated bus voltage
13. Steam Generator (SG) Water Level - Low Low	1, 2	3 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 11.3%

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.
(f) Above the P-8 (Power Range Neutron Flux) or P-7 (Low Power Reactor Trips Block) interlocks.

Table 3.3.1-1 (page 4 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
14. Turbine Trip					
a. Low Autostop Oil Pressure	1(g)	3	O	SR 3.3.1.10 SR 3.3.1.15	≥ 45 psig
b. Turbine Stop Valve Closure	1(g)	2	O	SR 3.3.1.10 SR 3.3.1.15	Closed
15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1, 2	2 trains	Q	SR 3.3.1.14	NA

(g) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 5 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
16. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2(d)	2	S	SR 3.3.1.11 SR 3.3.1.13	≥ 1.0E-10 amp
b. Low Power Reactor Trips Block, P-7					
1. Power Range Neutron Flux	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 12% RTP
2. Turbine Impulse Pressure	1	2	T	SR 3.3.1.7 SR 3.3.1.10	≤ 12% Full Load
c. Power Range Neutron Flux, P-8	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 11% RTP
d. Power Range Neutron Flux, P-9	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 12% RTP
e. Power Range Neutron Flux, P-10	1, 2	4	S	SR 3.3.1.11 SR 3.3.1.13	≥ 9% RTP
17. Reactor Trip Breakers(h) (RTBs)	1, 2	2 trains	R	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	2 trains	C	SR 3.3.1.4	NA

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(h) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 6 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1, 2	1 each per RTB	V	SR 3.3.1.4	NA
	3[a], 4[a], 5[a]	1 each per RTB	C	SR 3.3.1.4	NA
19. Automatic Trip Logic	1, 2	2 trains	Q	SR 3.3.1.5	NA
	3[a], 4[a], 5[a]	2 trains	C	SR 3.3.1.5	NA

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

Table 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value is defined by the following Trip Setpoint.

$$\Delta T \le \Delta T_0 \left\{ K_1 - K_2 (T - T') \left[\frac{(1 + \tau_1 s)}{(1 + \tau_2 s)} \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured Reactor Coolant System (RCS) ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP, = *°F.

P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, = * psig

$K_1 \le$ *
K_2 = */°F
K_3 = */psig
τ_1 = * sec
τ_2 = * sec

$f_1(\Delta I)$ = *{* + (q_t - q_b)}	when q_t - q_b $\le$ *% RTP
*% of RTP	when *% RTP < q_t - q_b $\le$ *% RTP
*{(q_t - q_b) - *}	when q_t - q_b > *% RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

* As specified in the COLR.

Table 3.3.1-1 (page 8 of 8)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value is defined by the following Trip Setpoint.

$$\Delta T \leq \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_3 s T}{1 + \tau_3 s} - K_6 \left(T - T' \right) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP, = *°F.

$K_4 \leq$ *

K_5 = */°F for increasing T_{avg}
= */°F for decreasing T_{avg}

K_6 = */°F when $T > T'$
= */°F when $T \leq T'$

τ_3 = * sec

* As specified in the COLR.

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s) or train(s).	Immediately
B. One channel or train inoperable.	B.1 Restore channel or train to OPERABLE status.	48 hours <u>OR</u> -------NOTE------- Not applicable to Function 2.a. ---------------------- In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One train inoperable.	-----------------NOTE---------------- One train may be bypassed for up to 8 hours for surveillance testing provided the other train is OPERABLE. ---	
	C.1 Restore train to OPERABLE status.	6 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
D. One channel inoperable.	-----------------NOTE---------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ---	
	D.1 Place channel in trip.	6 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One or more Containment Pressure channel(s) inoperable.	----------------NOTE---------------- One channel may be bypassed for up to 4 hours for surveillance testing. --	
	E.1.1 Place inoperable channel(s) in trip.	6 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
	<u>AND</u>	
	E.1.2 Verify one channel per pair OPERABLE.	6 hours
F. One channel or train inoperable.	F.1 Restore channel or train to OPERABLE status.	48 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. One train inoperable.	-----------------NOTE---------------- One train may be bypassed for up to 8 hours for surveillance testing provided the other train is OPERABLE. ---	
	G.1 Restore train to OPERABLE status.	6 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
H. One channel inoperable.	-----------------NOTE---------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ---	
	H.1 Place channel in trip.	6 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. One or both channel(s) inoperable on one bus.	-----------------NOTE---------------- One inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. --	
	I.1 Place channel(s) in trip.	6 hours <u>OR</u> -------NOTE------- Not applicable when more than one channel inoperable on one bus. ---------------------- In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
J. One train inoperable.	-----------------NOTE---------------- One train may be bypassed for up to 8 hours for surveillance testing provided the other train is OPERABLE. --- J.1 Enter applicable Condition(s) and Required Action(s) for Auxiliary Feedwater (AFW) train made inoperable by ESFAS instrumentation.	 Immediately
K. One channel inoperable.	K.1 Enter applicable Condition(s) and Required Action(s) for Auxiliary Feedwater (AFW) pump made inoperable by ESFAS instrumentation.	Immediately
L. Required Action and associated Completion Time of Conditions B or C not met.	L.1 Be in MODE 3. <u>AND</u> L.2 Be in MODE 5.	6 hours 36 hours
M. Required Action and associated Completion Time of Conditions D, E, F, or G not met.	M.1 Be in MODE 3. <u>AND</u> M.2 Be in MODE 4.	6 hours 12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
N. Required Action and associated Completion Time of Condition H or I not met.	N.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.2	Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.3	Perform COT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.4	---------------------------NOTE--------------------------- Verification of setpoint not required. --- Perform TADOT.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.5	----------------------------NOTE---------------------------- Verification of setpoint not required. --- Perform TADOT.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.6	----------------------------NOTE---------------------------- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. --- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.7	Perform MASTER RELAY TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.8	Perform SLAVE RELAY TEST.	In accordance with the Surveillance Frequency Control Program

Table 3.3.2-1 (page 1 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Safety Injection					
a. Manual Initiation	1, 2, 3, 4	2	B	SR 3.3.2.5	NA
b. Automatic Actuation Relay Logic	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.8	NA
c. High Containment Pressure	1, 2, 3	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≤ 4.0 psig
d. Pressurizer Low Pressure	1, 2, 3[(a)]	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≥ 1760 psig
e. Steam Line Low Pressure	1, 2, 3[(a)]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≥ 500[(b)] psig
2. Containment Spray					
a. Manual Initiation	1, 2, 3, 4	2	B	SR 3.3.2.4	NA
b. Automatic Actuation Relay Logic	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.8	NA

(a) Pressurizer Pressure ≥ 2000 psig.
(b) Time constants used in the lead/lag controller are $t_1 \geq 12$ seconds and $t_2 \leq 2$ seconds.

Table 3.3.2-1 (page 2 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Containment Spray (continued)					
c. High-High Containment Pressure	1, 2, 3	3 sets of 2	E	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≤ 23 psig
3. Containment Isolation					
a. Manual Initiation	1, 2, 3, 4	2	B	SR 3.3.2.4	NA
b. Automatic Actuation Relay Logic	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.8	NA
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
4. Steam Line Isolation					
a. Manual Initiation	1, 2[(c)], 3[(c)]	1/loop	F	SR 3.3.2.4	NA
b. Automatic Actuation Relay Logic	1, 2[(c)], 3[(c)]	2 trains	G	SR 3.3.2.2 SR 3.3.2.7	NA
c. High-High Containment Pressure	1, 2[(c)], 3[(c)]	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≤ 17 psig

(c) Except when both Main Steam Isolation Valves (MSIVs) are closed.

Table 3.3.2-1 (page 3 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Steam Line Isolation (continued)					
d. High Steam Flow	1, 2$^{(c)}$, 3$^{(c)(d)}$	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≤ 9.18E5 lb/hr at 1005 psig
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
and					
Coincident with Low-Low T_{avg}	1, 2$^{(c)}$, 3$^{(c)(d)}$	4	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≥ 536°F
e. High High Steam Flow	1, 2$^{(c)}$, 3$^{(c)}$	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≤ 4.5E6 lb/hr at 735 psig
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
5. Feedwater Isolation					
a. Automatic Actuation Relay Logic	1 ,2$^{(e)}$, 3$^{(e)}$	2 trains	G	SR 3.3.2.2 SR 3.3.2.7	NA
b. High- High Steam Generator (SG) Water Level	1, 2$^{(e)}$	3 per SG	H	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≤ 90%

(c) Except when both MSIVs are closed.
(d) Reactor Coolant System (RCS) $T_{avg} \geq 520°F$
(e) Except when all Main Feedwater Regulation Valves (MFRVs) and MFRV bypass valves are closed and de-activated or isolated by a closed manual valve.

Table 3.3.2-1 (page 4 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Feedwater Isolation (continued)					
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
6. Auxiliary Feedwater					
a. Automatic Actuation Relay Logic	1, 2, 3	2 trains	J	SR 3.3.2.2	NA
b. Low-Low SG Water Level	1, 2, 3	3 per SG	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≥ 11.3%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
d. Undervoltage on 4 kV Buses 11 and 12 (21 and 22)(f)	1, 2	2 per bus	I	SR 3.3.2.4 SR 3.3.2.6	≥ 76% rated bus voltage
e. Trip of both Main Feedwater Pumps	1, 2(g)	2 per pump	K	SR 3.3.2.4	NA

(f) Start of Turbine Driven Pump only.
(g) This Function may be bypassed during alignment and operation of the AFW System for SG level control.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTE----------- Only applicable in MODE 1, 2, 3, or 4. -------------------------------	C.1 Perform SR 3.3.4.2 for OPERABLE automatic load sequencer.	6 hours <u>AND</u> Once per 24 hours thereafter
Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Function a or b or both with three channels per bus inoperable. <u>OR</u> One required automatic load sequencer inoperable.	<u>AND</u> C.2 Establish offsite paths block loading capability for associated 4 kV safeguards bus. <u>AND</u> C.3 Verify offsite paths for associated 4kV safeguards bus OPERABLE. <u>AND</u>	 8 hours 8 hours <u>AND</u> Once per 8 hours thereafter

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.4 Declare required feature(s) supported by the affected inoperable DG inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND	
	C.5 Restore automatic load sequencer to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. -----------NOTE----------- Only applicable in MODES 5 or 6. ------------------------------ Required Action and associated Completion Time of Condition A or B not met. OR Function a or b or both with three channels per bus inoperable. OR One required automatic load sequencer inoperable.	E.1 Enter applicable Condition(s) and Required Action(s) of LCO 3.8.2, “AC Sources – Shutdown” for the DG made inoperable from inoperable 4 kV safeguards bus voltage instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform COT on each undervoltage and degraded voltage channel.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.2	Perform ACTUATION LOGIC TEST on each automatic load sequencer.	In accordance with the Surveillance Frequency Control Program
SR 3.3.4.3	Perform CHANNEL CALIBRATION on undervoltage and degraded voltage channels with Allowable Value as follows: a. Undervoltage Allowable Value $\geq$ 3016 V and $\leq$ 3224 V with an undervoltage time delay of 4 ± 1.5 seconds. b. Degraded voltage Allowable Value $\geq$ 3944 V and $\leq$ 4002 V with a degraded voltage time delay of 8 ± 0.5 seconds and degraded voltage DG start time delay of 7.5 to 63 seconds.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One group of pressurizer heaters inoperable.	B.1 Restore group of pressurizer heaters to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is $\leq$ 90%.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters is $\geq$ 100 kW.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.3	Verify required pressurizer heaters are capable of being powered from an emergency power supply.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One PORV inoperable and not capable of being manually cycled.	B.1 Close associated block valve.	1 hour
	AND	
	B.2 Remove power from associated block valve.	1 hour
	AND	
	B.3 Restore PORV to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
C. One block valve inoperable.	-----------------NOTE---------------- Required Actions C.1 and C.2 do not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.2 ---	
	C.1 Place associated PORV in manual control.	1 hour
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Restore block valve to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours
E. Both PORVs inoperable and not capable of being manually cycled.	E.1 Close associated block valves. <u>AND</u> E.2 Remove power from associated block valves. <u>AND</u> E.3 Be in MODE 3. <u>AND</u> E.4 Be in MODE 4.	1 hour 1 hour 6 hours 12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Both block valves inoperable.	-----------------NOTE---------------- Required Action F.1 does not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.2 ---	
	F.1 Restore one block valve to OPERABLE status.	2 hours
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	G.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	---------------------------NOTES-------------------------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each block valve.	 In accordance with the Surveillance Frequency Control Program
SR 3.4.11.2	---------------------------NOTE--------------------------- Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each PORV.	 In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS – Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

-------------------------------------NOTE---------------------------------------
In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.15.1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.2.1 Verify the following valves are in the listed position. Unit 1 Valve Number / Westinghouse Valve Number / Position / Function 32070 / 8801A / OPEN / SI Injection to RCS Cold Leg A 32068 / 8801B / OPEN / SI Injection to RCS Cold Leg B 32073 / 8806A / OPEN / SI Cold Leg Injection Line 32206 / 8816A / CLOSED / SI Pump Suction from RHR 32207 / 8816B / CLOSED / SI Pump Suction from RHR Unit 2 Valve Number / Westinghouse Valve Number / Position / Function 32173 / 8801A / OPEN / SI Injection to RCS Cold Leg A 32171 / 8801B / OPEN / SI Injection to RCS Cold Leg B 32176 / 8806A / OPEN / SI Cold Leg Injection Line 32208 / 8816A / CLOSED / SI Pump Suction from RHR 32209 / 8816B / CLOSED / SI Pump Suction from RHR	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.2	---------------------------NOTE--------------------------- Not required to be met for system vent flow paths opened under administrative control. --- Verify each ECCS manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.3	Verify power to the valve operator has been removed for each valve listed in SR 3.5.2.1.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.4	Verify ECCS accessible locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.5	Verify ECCS inaccessible locations susceptible to gas accumulation are sufficiently filled with water.	Prior to entering MODE 3 after exiting shutdown cooling
SR 3.5.2.6	Verify each ECCS pump's developed heat at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.7	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.8	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.9	Verify each ECCS throttle valve listed below is in the correct position. Unit 1 Valve Number / Unit 2 Valve Number SI-15-6 / 2SI-15-6 SI-15-7 / 2SI-15-7 SI-15-8 / 2SI-15-8 SI-15-9 / 2SI-15-9	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.10	Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	<u>AND</u>	
	C.2 Verify a door is closed in the affected air lock.	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.	6 hours
	<u>AND</u>	
	D.2 Be in MODE 5.	36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE----------- Only applicable to penetration flow paths with two containment isolation valves. ------------------------------- One or more penetration flow paths with one containment isolation valve inoperable for reasons other than Condition D.	A.1 Isolate the affected penetration flow paths by use of at least one closed and de-activated or mechanically blocked power operated valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
	<u>AND</u>	
	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 paths is isolated.	Once per 31 days following isolation for isolation devices outside containment <u>AND</u>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTE----------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. ------------------------------ One or more penetration flow paths with one containment isolation valve inoperable.	C.1 Isolate the affected penetration flow paths by use of at least one closed and de-activated power operated valve, closed manual valve, or blind flange.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
	<u>AND</u>	
	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 paths is isolated.	Once per 31 days following isolation

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Spray and Cooling Systems

LCO 3.6.5 Two containment spray trains and two containment cooling trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 84 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or both containment cooling fan coil unit(s) (FCU) in one train inoperable.	C.1 Restore containment cooling FCU(s) to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
D. One containment cooling FCU in each train inoperable.	D.1 Initiate action to isolate both inoperable FCUs.	Immediately
	<u>AND</u>	
	D.2 Restore all FCUs to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	E.2 Be in MODE 5.	36 hours

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	8 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.2.1 ----------------------------NOTE---------------------------- Only required to be performed in MODES 1 and 2. --- Verify the isolation time of each MSIV is within limits.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.2.2	----------------------------NOTE---------------------------- Only required to be performed in MODES 1 and 2. --- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.4 Steam Generator (SG) Power Operated Relief Valves (PORVs)

LCO 3.7.4 Two SG PORV lines shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SG PORV line inoperable.	A.1 Restore SG PORV line to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. Two SG PORV lines inoperable.	B.1 Restore one SG PORV line to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours 12 hours

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable. <u>OR</u> ------------NOTE----------- Only applicable if MODE 2 has not been entered following refueling. ------------------------------- One turbine driven AFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. One AFW train inoperable in MODE 1, 2, or 3 for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CC) System

LCO 3.7.7 Two CC trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CC train inoperable.	A.1 -------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CC. --------------------------------- Restore CC train 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.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	----------------------------NOTE---------------------------- Isolation of CC flow to individual components does not render the CC System inoperable. --- Verify each CC manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.2	----------------------------NOTE---------------------------- This SR only applies to those valves required to align CC System to support the safety injection or recirculation phase of emergency core cooling. --- Verify each CC automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.3	Verify each CC pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.8 Cooling Water (CL) System

LCO 3.7.8 Two CL trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No safeguards CL pumps OPERABLE for one train.	-----------------NOTES---------------- 1. Unit 1 enter applicable Conditions and Required Actions of LCO 3.8.1, “AC Sources-MODES 1, 2, 3, and 4,” for emergency diesel generator made inoperable by CL System. 2. Both units enter applicable Conditions and Required Actions of LCO 3.4.6, “RCS Loops-MODE 4,” for residual heat removal loops made inoperable by CL System. --	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Restore one safeguards CL pump to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. One CL supply header inoperable.	----------------NOTES--------------- 1. Unit 1 enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-MODES 1, 2, 3, and 4," for emergency diesel generator made inoperable by CL System. 2. Both units enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4," for residual heat removal loops made inoperable by CL System. --- B.1 Verify vertical motor driven CL pump OPERABLE. <u>AND</u>	 4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Verify opposite train diesel driven CL pump OPERABLE. AND B.3 Restore CL supply header to OPERABLE status.	4 hours 72 hours OR In accordance with the Risk Informed Completion Time Program
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. -----------NOTE------------ Separate Condition entry is allowed for each stored diesel driven CL pump fuel oil supply. ------------------------------- One or both stored diesel driven CL pump fuel oil supply(s) < 7 days and ≥ 6 days.	D.1 Restore fuel oil supply to ≥ 7 days.	48 hours

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources-Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. Two paths between the offsite transmission grid and the onsite 4 kV Safeguards Distribution System; and

b. Two diesel generators (DGs) capable of supplying the onsite 4 kV Safeguards Distribution System.

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

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required path inoperable.	A.1 Perform SR 3.8.1.1 for the OPERABLE path. AND	1 hour AND Once per 8 hours thereafter

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Restore path to OPERABLE status.	7 days <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. One DG inoperable.	B.1	Perform SR 3.8.1.1 for the paths.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		
	B.2	Declare required feature(s) supported by the inoperable DG inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure. OR B.3.2 Perform SR 3.8.1.2 for OPERABLE DG. AND B.4 Restore DG to OPERABLE status.	24 hours 24 hours 14 days OR In accordance with the Risk Informed Completion Time Program
C. Two paths inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. AND	12 hours from discovery of Condition C concurrent with inoperability of redundant required features

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Restore one path to OPERABLE status.	24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
D. One path inoperable. <u>AND</u> One DG inoperable.	-----------------NOTE---------------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems-Operating," when Condition D is entered with no AC power source to either train. --- D.1 Restore path to OPERABLE status. <u>OR</u>	 12 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	D.2 Restore DG to OPERABLE status.	12 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
E. Two DGs inoperable.	E.1 Restore one DG 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. <u>AND</u> F.2 Be in MODE 5.	6 hours 36 hours
G. Two DGs inoperable and one or more paths inoperable. <u>OR</u> One DG inoperable and two paths inoperable.	G.1 Enter LCO 3.0.3.	Immediately

*A one-time change increased the Completion Time to 12 hours for Unit 2 during the period from January 29 through January 31, 2019. This change was approved via an emergency license amendment

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required path.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.2	-----------------------------NOTES------------------------- 1. Performance of SR 3.8.1.6 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR in consideration of manufacturer's recommendations. When modified start procedures are not used, the time, voltage, and frequency tolerances of SR 3.8.1.6 must be met. --- Verify each DG starts from standby conditions and achieves steady state voltage $\geq$ 4084 V and $\leq$ 4400 V, and frequency $\geq$ 59.5 Hz and $\leq$ 60.5 Hz.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	---------------------------NOTES--------------------------- 1. DG loadings may include gradual loading in consideration of manufacturer's recommendations. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.6. --- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load: a. Unit 1; ≥ 2500 kW; and b. Unit 2; ≥ 5100 kW and ≤ 5300 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.4	Verify fuel oil level above lower limit switch in each day 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 storage tank to the day tank.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.6	------------------------NOTE------------------------ All DG starts may be preceded by an engine prelube period. -- Verify each DG starts from standby condition and achieves: a. In ≤ 10 seconds, voltage ≥ 3740 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 4084 V and ≤ 4400 V, and frequency ≥ 59.5 Hz and ≤ 60.5 Hz.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.7	Verify each DG does not trip during and following a load rejection of: 1. Unit 1 ≥ 650 kW; and 2. Unit 2 ≥ 860 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.8	Verify each DG's automatic trips are bypassed on an actual or simulated safety injection signal except: a. Engine overspeed; b. Generator differential current; and c. Ground fault (Unit 1 only).	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	---------------------------NOTES------------------------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.85. 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 DG operates for ≥ 24 hours: a. For ≥ 2 hours loaded: Unit 1 ≥ 2832 kW, and ≤ 3000 kW Unit 2 ≥ 5400 kW, and ≤ 5940 kW; and b. For the remaining hours of the test loaded: Unit 1 ≥ 2500 kW, and Unit 2 ≥ 4860 kW; and c. Achieves steady state voltage ≥ 4084 V and ≤ 4400 V; and frequency ≥ 59.5 Hz and ≤ 60.5 Hz.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	-----------------------------NOTES------------------------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 3. 12 Battery Charger not required to be energized in SR 3.8.1.10(c) until completion of this SR during Unit 1 2011 refueling outage.* --- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated safety injection actuation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and energizes emergency loads in ≤ 60 seconds.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.11	-----------------------------NOTE--------------------------- All DG starts may be preceded by an engine prelube period. --- Verify on an actual or simulated loss of offsite power signal that the DG auto-starts from standby condition.	In accordance with the Surveillance Frequency Control Program

*A modification will be installed during or prior to the Unit 1 2011 refueling outage to assure the 12 Battery Charger is automatically powered from its normal bus within 60 seconds. Compliance with this SR will be demonstrated after implementation of the modification.

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One battery charger inoperable.	A.1 Verify its associated battery is OPERABLE.	2 hours
	AND	
	A.2 Verify the other train battery charger is OPERABLE.	2 hours
	AND	
	A.3 Verify the diesel generator and safeguards equipment on the other train are OPERABLE.	2 hours
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Restore battery charger to OPERABLE status.	8 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. One battery inoperable.	B.1 Verify associated battery charger is OPERABLE.	2 hours
	<u>AND</u>	
	B.2 Verify other train battery is OPERABLE.	2 hours
	<u>AND</u>	
	B.3 Verify other train battery charger is OPERABLE.	2 hours
	<u>AND</u>	
	B.4 Restore battery to OPERABLE status.	8 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One DC electrical power subsystem inoperable for reasons other than Condition A or B.	C.1 Restore DC electrical power subsystem to OPERABLE status.	2 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. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.2	Verify each battery charger supplies ≥ 250 amps at greater than or equal to the minimum established float voltage for ≥ 4 hours. <u>OR</u> Verify each battery charger can recharge the battery to the fully charged state within 24 hours while supplying the demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	-------------------------NOTES--------------------------- 1. The modified performance discharge test in SR 3.8.6.6 may be performed in lieu of SR 3.8.4.3. 2. This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters-Operating

LCO 3.8.7 Four Reactor Protection Instrument AC inverters shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One Reactor Protection Instrument AC inverter inoperable.	A.1	------------NOTE------------ Enter the applicable Conditions and Required Actions of LCO 3.8.9, “Distribution Systems – Operating” with any Reactor Protection Instrument AC panel de-energized. -------------------------------- Restore Reactor Protection Instrument AC inverter to OPERABLE status.	24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage and alignment to required Reactor Protection Instrument AC panels.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems-Operating

LCO 3.8.9 Train A and Train B safeguards AC and DC, and Reactor Protection Instrument AC electrical power distribution subsystems shall be OPERABLE.

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

ACTIONS

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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more safeguards DC electrical power distribution subsystems inoperable.	B.1 Restore safeguards DC electrical power distribution subsystems to OPERABLE status.	2 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
C. One Reactor Protection Instrument AC panel inoperable.	C.1 Restore Reactor Protection Instrument AC panel to OPERABLE status.	2 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. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two trains with inoperable distribution subsystems that result in a loss of safety function. <u>OR</u> Two or more Reactor Protection Instrument AC panels inoperable.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker and switch alignments and voltage to safeguards AC, DC, and Reactor Protection Instrument AC electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

5.5 Programs and Manuals (continued)

5.5.9 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of the Control Room Special Ventilation System (CRSVS), Auxiliary Building Special Ventilation System (ABSVS), and Shield Building Ventilation System (SBVS) at least once each 24 months.

Demonstrate for the ABSVS, SBVS, and CRSVS systems that:

a. An inplace DOP test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% (for DOP, particles having a mean diameter of 0.7 microns);

b. A halogenated hydrocarbon test of the inplace charcoal adsorber shows a penetration and system bypass < 0.05% (SBVS not applicable);

c. A laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than: 1) 10% penetration for ABSVS, and 2) 2.5% penetration for the CRSVS when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and 95% relative humidity (RH);

d. The pressure drop across the combined HEPA filters and the charcoal adsorbers (SBVS not applicable to charcoal adsorbers) is less than 6 inches of water at the system flowrate ± 10%; and

e. A laboratory test of a sample of the charcoal adsorber shall have filter test face velocities greater than or equal to the following values for each system: 1) 54 fpm for the CRSVS, and 2) 72 fpm for the ABSVS.

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas holdup system, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of oxygen in the waste gas holdup system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria;

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than or equal to 78,800 Curies of noble gas (considered as dose equivalent Xe-133); and

c. A surveillance program to ensure that the quantity of radioactivity contained in each of the following tanks shall be limited to 10 Curies, excluding tritium and dissolved or entrained noble gases:

   Condensate storage tanks
   Outside temporary tanks

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance Frequencies.

5.5 Programs and Manuals (continued)

5.5.11 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with the limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment. Acceptability of new fuel oil shall be determined prior to addition to the safeguards storage tanks. Testing of diesel fuel oil stored in the safeguards storage tanks shall be performed at least every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies.

5.5.12 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews;

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. a change in the TS incorporated in the license, or

   2. a change to the USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59;

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR; and

5.5 Programs and Manuals

5.5.12 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of Specification 5.5.12 b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with USAR updates.

5.5.13 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the inoperable support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5 Programs and Manuals (continued)

5.5.14 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exception:

   1. Unit 1 and Unit 2 (steam generator (SG) replacement commencing Fall 2013) are excepted from post-modification integrated leakage rate testing requirements associated with SG replacement.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident is less than the containment internal design pressure, P_a, of 46 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.15% of primary containment air weight per day. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage shall be less than 0.06% of primary containment air weight per day at pressure P_a. For pipes connected to systems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.006% of primary containment air weight per day at pressure P_a.

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program (continued)

d. Leakage Rate acceptance criteria are:

1. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to unit startup, following testing in accordance with the program, the combined leakage rate acceptance criteria are $\leq 0.60\ L_a$ for all components subject to Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at ≥ 46 psig.

b) For each door intergasket test, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.15 Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance of the 125V plant safeguards batteries and service building batteries, which may be used instead of the safeguards batteries during shutdown conditions in accordance with manufacturer's recommendations, as follows:

a. Actions to restore battery cells with float voltage < 2.13 V will be in accordance with manufacturer's recommendations, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

5.5 Programs and Manuals (continued)

5.5.16 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Special Ventilation System (CRSVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design conditions including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air in-leakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Licensee controlled programs that will be used to verify the integrity of the CRE boundary. Conditions that generate relevant information from those programs will be entered into the corrective action process and shall be trended and used as part of the periodic assessments of the CRE boundary.

5.5 Programs and Manuals

5.5.16 Control Room Envelope Habitability Program (continued)

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

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability and determining CRE unfiltered in-leakage as required by paragraph c.

5.5.17 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 (continued)

5.5.18 Risk Informed Completion Time Program

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

a. The RICT may not exceed 30 days;

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

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

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

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

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

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

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

5.5 Programs and Manuals

5.5.18 Risk Informed Completion Time Program (continued)

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 235 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 223 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY - MINNESOTA

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By application dated December 16, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19350C188), as supplemented by letter dated September 1, 2020 (ADAMS Accession No. ML20245E401), Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), submitted a license amendment request (LAR) for Prairie Island Nuclear Generating Plant, Units 1 and 2 (Prairie Island).

The supplemental letter dated September 1, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 25, 2020 (85 FR 10733).

The proposed 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 NRC issued a final model safety evaluation (SE) approving TSTF-505, Revision 2, on November 21, 2018 (ADAMS Accession No. ML18269A041).

2.0 REGULATORY EVALUATION

2.1 Description of Risk-Informed Completion Time 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., shut down the reactor).

The Nuclear Energy Institute (NEI) Topical Report 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. ML12286A322), provides a methodology for extending existing CTs and thereby delay 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 licensee’s December 16, 2019, LAR, as supplemented by letter dated September 1, 2020, requested approval to add a RICT Program to the Administrative Controls section of the TSs, add new conditions and associated actions in some TSs, and 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 1.3, "Completion Times"

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

EXAMPLE 1.3-8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

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

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

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

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

2.2.2 TS 5.5.18, "Risk-Informed Completion Time Program"

TS 5.5.18, which describes the RICT Program, would be added to the TSs 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 MODES 1 and 2;

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

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

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

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

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

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

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the 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 would be 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 <u>*OR*</u> *In accordance with the Risk Informed Completion Time Program*

Where necessary, conforming changes would be 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 that would be modified by the proposed change are identified below.

<u>TS 3.3.1, "Reactor Trip System (RTS) Instrumentation"</u>

- Required Action B.1
    - The option of calculating a RICT is applied to the action to restore channel to OPERABLE status (for Condition B, One Manual Reactor Trip channel inoperable).

- Required Action B.2
    - The option to be in MODE 3 is deleted (addressed by new Condition W for when Required Action and associated CT not met).

- Required Action D.1.1
    - The option of calculating a RICT is applied to the action to place channel in trip (for Condition D, One Power Range Neutron Flux channel inoperable).

- Required Action D.2
    - The option to be in MODE 3 is deleted (addressed by new Condition W for when Required Action and associated CT not met).

- Required Action E.1
    - Place channel in trip (for Condition E, One channel inoperable).

- Required Action E.2
    - The option to be in MODE 3 is deleted (addressed by new Condition W for when Required Action and associated CT not met).

- Required Action K.1
    - Place channel in trip (for Condition K, One channel inoperable).

- Required Action K.2
    - The option to reduce thermal power to <P-7 and P-8 is deleted (addressed by new Condition N for when Required Action and associated CT not met).

- Required Action L.2
    - The option to reduce thermal power to <P-7 and P-8 is deleted (addressed by new Condition N for when Required Action and associated CT not met).

- Required Action M.2
    - The option to reduce thermal power to <P-7 and P-8 is deleted (addressed by new Condition N for when Required Action and associated CT not met).

- New Condition N is added: to reduce THERMAL POWER to < P-7 and P-8 in 6 hours when Required Action and associated CT of Conditions K, L, or M not met.

- Required Action O.1 (previously N.1 before insertion of new Conditions N and P)
    - Place channel in trip (for Condition O, One Turbine Trip channel inoperable).

- Required Action O.2 (previously N.2 before insertion of new Conditions N and P)
    - The option to be in MODE 3 is deleted (addressed by new Condition W for when Required Action and associated CT not met).

- New Condition P is added: to reduce THERMAL POWER to < P-9 in 6 hours when Required Action and associated CT of Condition O not met.

- Required Action Q.1 (previously O.1 before insertion of new Conditions N and P)
    - Restore train to OPERABLE status (for Condition Q, One train inoperable).

- Required Action Q.2 (previously O.2 before insertion of new Conditions N and P)
  - The option to be in MODE 3 is deleted (addressed by new Condition W for when Required Action and associated CT not met).

- Required Action R.1 (previously P.1 before insertion of new Conditions N and P)
  - Restore train to OPERABLE status (for Condition R, One Reactor Trip Breaker train inoperable).

- Required Action R.2 (previously P.2 before insertion of new Conditions N and P)
  - The option to be in MODE 3 is deleted (addressed by new Condition W for when Required Action and associated CT not met).

- Required Action S.1 (previously Q.1 before insertion of new Conditions N and P)
  - Renumbered (for Condition S, One or more channels inoperable).

- Required Action S.2 (previously Q.2 before insertion of new Conditions N and P)
  - The option to be in MODE 3 is deleted (addressed by new Condition W for when Required Action and associated CT not met).

- Required Action T.1 (previously R.1 before insertion of new Conditions N and P)
  - Renumbered (for Condition T, One or more channels inoperable).

- Required Action T.2 (previously R.2 before insertion of new Conditions N and P)
  - The option to be in MODE 2 is deleted (addressed by new Condition W for when Required Action and associated CT not met).

- New Condition U is added: to be in MODE 2 in 6 hours when Required Action and associated CT of Condition T is not met.

- Required Action V.1 (previously S.1 before insertion of new Conditions N, P, and U)
  - Renumbered (for Condition V, One trip mechanism inoperable for one Reactor Trip Breaker).

- Required Action V.2 (previously S.2 before insertion of new Conditions N, P, and U)
  - The option to be in MODE 3 is deleted (addressed by new Condition W for when Required Action and associated CT not met).

- New Condition W is added: to be in MODE 3 in 6 hours when Required Action and associated CT of Conditions B, D, E, Q, R, S, or V is not met.

- Table 3.3.1-1, RTS Instrumentation
  - Conditions in Table 3.3.1-1 are relabeled consistent with the above changes.

<u>TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation"</u>

- Required Action B.2.1 and B.2.2
  - The option to be in MODE 3 and then in MODE 5 is deleted (addressed by new Condition L for when Required Action and associated CT not met).

- Required Action C.1
    - The option of calculating a RICT is applied to the action to restore train to OPERABLE status (for Condition C, One train inoperable).
- Required Action C.2.1 and C.2.2
    - The option to be in MODE 3 and then in MODE 5 is deleted (addressed by new Condition L for when Required Action and associated CT not met).
- Required Action D.1
    - The option of calculating a RICT is applied to the action to place channel in trip (for Condition D, One channel inoperable).
- Required Action D.2.1 and D.2.2
    - The option to be in MODE 3 and then in MODE 4 is deleted (addressed by new Condition M for when Required Action and associated CT not met).
- Required Action F.1
    - The option of calculating a RICT is applied to the action to restore channel or train to OPERABLE status (for Condition F, One channel or train inoperable).
- Required Action F.2.1 and F.2.2
    - The option to be in MODE 3 and then in MODE 4 is deleted (addressed by new Condition M for when Required Action and associated CT not met).
- Required Action G.1
    - The option of calculating a RICT is applied to the action to restore train to OPERABLE status (for Condition G, One train inoperable).
- Required Action G.2.1 and G.2.2
    - The option to be in MODE 3 and then in MODE 4 is deleted (addressed by new Condition M for when Required Action and associated CT not met).
- Required Action H.1
    - The option of calculating a RICT is applied to the action to place channel in trip (for Condition H, One channel inoperable).
- Required Action H.2
    - The option to be in MODE 3 is deleted (addressed by new Condition N for when Required Action and associated CT not met).
- New Condition L is added: to be in MODE 3 in 6 hours and MODE 5 in 36 hours when Required Action and associated CT of Conditions B or C is not met.
- New Condition M is added: to be in MODE 3 in 6 hours and MODE 4 in 12 hours when Required Action and associated CT of Conditions D, E, F, or G is not met.
- New Condition N is added: to be in MODE 3 in 6 hours when Required Action and associated CT of Condition H or I is not met.

<u>TS 3.3.4, “4 kV Safeguards Bus Voltage Instrumentation”</u>

- Required Action C.5
    - The option of calculating a RICT is applied to the action to restore automatic load sequencer to OPERABLE status (for Condition C, Required Action and associated completion time of Condition A or B not met, or Function a or b or both with three channels per bus inoperable, or one required automatic load sequencer inoperable).

<u>TS 3.4.9, “Pressurizer”</u>

- Required Action B.1
    - The option of calculating a RICT is applied to the action to restore group of pressurizer heaters to OPERABLE status (for Condition B, One group of pressurizer heaters inoperable).

<u>TS 3.4.11, “Pressurizer Power Operated Relief Valves (PORVs)”</u>

- Required Action B.3
    - The option of calculating a RICT is applied to the action to restore PORV to OPERABLE status (for Condition B, One PORV inoperable and not capable of being manually cycled).

- Required Action C.2
    - The option of calculating a RICT is applied to the action to restore block valve to OPERABLE status (for Condition C, One block valve inoperable).

<u>TS 3.5.2, “ECCS [Emergency Core Cooling System] – Operating”</u>

- Required Action A.1
    - The option of calculating a RICT is applied to the action to restore train(s) to OPERABLE status (for Condition A, One or more trains inoperable).

<u>TS 3.6.2, “Containment Air Locks”</u>

- Required Action C.3
    - The option of calculating a RICT is applied to the action to restore air lock to OPERABLE status (for Condition C, One or more containment air locks inoperable for reasons other than Condition A or B).

<u>TS 3.6.3, “Containment Isolation Valves”</u>

- Required Action A.1
    - The option of calculating a RICT is applied to the action to isolate the affected penetration flow paths by use of at least one closed and de-activated or mechanically blocked power operated valve, closed manual valve, blind flange, or check valve with flow through the valve secured (for Condition A, One or more penetration flow paths with one containment isolation valve inoperable for reasons other than Condition D).

- Required Action C.1
  - The option of calculating a RICT is applied to the action to isolate the affected penetration flow paths by use of at least one closed and de-activated power operated valve, closed manual valve, or blind flange (for Condition C, One or more penetration flow paths with one containment isolation valve inoperable).

- Required Actions A.2 and C.2
  - The CTs for the Required Actions to verify the affected penetration flow path is isolated, have been modified by adding the words “following isolation” after “Once per 31 days.”

<u>TS 3.6.5, “Containment Spray and Cooling Systems”</u>

- Required Action A.1
  - The option of calculating a RICT is applied to the action to restore containment spray train to OPERABLE status (for Condition A, One containment spray train inoperable).

- Required Action C.1
  - The option of calculating a RICT is applied to the action to restore containment cooling [fan coil unit(s)] FCU(s) to OPERABLE status (for Condition C, One or both containment cooling fan coil unit(s) in one train inoperable).

<u>TS 3.7.2, “Main Steam Isolation Valves (MSIVs)”</u>

- Required Action A.1
  - The option of calculating a RICT is applied to the action to restore MSIV to OPERABLE status (for Condition A, One MSIV inoperable in MODE 1).

<u>TS 3.7.4, “Steam Generator (SG) Power Operated Relief Valves (PORVs)”</u>

- Required Action A.1
  - The option of calculating a RICT is applied to the action to restore SG PORV to OPERABLE status (for Condition A, One SG PORV line inoperable).

<u>TS 3.7.5, “Auxiliary Feedwater (AFW) System”</u>

- Required Action A.1
  - The option of calculating a RICT is applied to the action to restore affected equipment to OPERABLE status (for Condition A, One steam supply to turbine driven AFW pump inoperable or one turbine driven AFW pump inoperable in MODE 3 following refueling).

- Required Action B.1
  - The option of calculating a RICT is applied to the action to restore AFW train to OPERABLE status (for Condition B, One AFW train inoperable in MODE 1, 2, or 3 for reasons other than Condition A).

<u>TS 3.7.7, “Component Cooling Water (CC) System”</u>

- Required Action A.1
    - The option of calculating a RICT is applied to the action to restore Component Cooling train to OPERABLE status (for Condition A, One Component Cooling train inoperable).

<u>TS 3.8.1, “AC [Alternating Current] Sources – Operating”</u>

- Required Action A.2
    - The option of calculating a RICT is applied to the action to restore path to OPERABLE status (for Condition A, One required path inoperable).

- Required Action B.4
    - The option of calculating a RICT is applied to the action to restore DG [diesel generator] to OPERABLE status (for Condition B, One DG inoperable).

- Required Action C.2
    - The option of calculating a RICT is applied to the action to restore one path to OPERABLE status (for Condition C, Two paths inoperable).

- Required Action D.1
    - The option of calculating a RICT is applied to the action to restore path to OPERABLE status (for Condition D, One path inoperable and one DG inoperable).

- Required Action D.2
    - The option of calculating a RICT is applied to the action to restore DG to OPERABLE status (for Condition D, One path inoperable and one DG inoperable).

<u>TS 3.8.4, “DC [Direct Current] Sources – Operating”</u>

- Required Action A.4
    - The option of calculating a RICT is applied to the action to restore battery charger to OPERABLE status (for Condition A, One battery charger inoperable).

- Required Action B.4
    - The option of calculating a RICT is applied to the action to restore battery to OPERABLE status (for Condition B, One battery inoperable).

- Required Action C.1
    - The option of calculating a RICT is applied to the action to restore DC electrical power subsystem to OPERABLE status (for Condition C, One DC electrical power subsystem inoperable for reasons other than Condition A or B).

TS 3.8.7, “Inverters-Operating”

- Required Action A.1
  - The option of calculating a RICT is applied to the action to restore Reactor Protection Instrument AC inverter to OPERABLE status (for Condition A, One Reactor Protection Instrument AC inverter inoperable).

TS 3.8.9, “Distribution Systems-Operating”

- Required Action A.1
  - The option of calculating a RICT is applied to the action to restore safeguards AC electrical power distribution subsystems to OPERABLE status (for Condition A, One or more safeguards AC electrical power distribution subsystems inoperable).

- Required Action B.1
  - The option of calculating a RICT is applied to the action to restore safeguards DC electrical power distribution subsystems to OPERABLE status (for Condition B, One or more safeguards DC electrical power distribution subsystems inoperable).

- Required Action C.1
  - The option of calculating a RICT is applied to the action to restore Reactor Protection Instrument AC panel to OPERABLE status (for Condition C, One Reactor Protection Instrument AC panel inoperable).

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 would be modified to permit the application of a RICT.

TS 3.3.1, “Reactor Trip System (RTS) Instrumentation”

- Required Action L.1
  - The option of calculating a RICT is applied to the action to place channel(s) in trip. A CT note is inserted that RICT entry is not permitted for the loss of function condition when more than one channel is inoperable on more than one bus (for Condition L, One or both channel(s) inoperable on one bus).

- Required Action M.1
  - The option of calculating a RICT is applied to the action to restore channel to OPERABLE status. A CT note is inserted that RICT entry is not applicable when THERMAL POWER is below P-8 and above P-7 (for Condition M, One Reactor Coolant Pump Breaker Open channel inoperable).

TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation"

- Required Action B.1
  - The option of calculating a RICT is applied to the action to restore channel or train to OPERABLE status. A CT note is inserted that RICT entry is not applicable to Function 2.a. (for Condition B, One channel or train inoperable).

TS 3.7.8, "Cooling Water (CL) System"

- Required Action B.3
  - The option of calculating a RICT is applied to the action to restore CL supply header to OPERABLE status (for Condition B, One CL supply header inoperable).

2.2.4.2 Application of the RICT to Additional ACTIONS Requirements

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

TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation"

- Required Action E.1.1
  - The option of calculating a RICT is applied to the action to place inoperable channel(s) in trip (for Condition E, One or more Containment Pressure channel(s) inoperable).

- Required Action E.2.1 and E.2.2
  - The option to be in MODE 3 and then in MODE 4 is deleted (addressed by new Condition M for when Required Action and associated CT not met).

- Required Action I.1
  - The option of calculating a RICT is applied to the action to place channel(s) in trip. A CT note is inserted that RICT entry is not permitted for the loss of function condition when more than one channel is inoperable on more than one bus (for Condition I, One or both channel(s) inoperable on one bus).

- Required Action I.2
  - The option to be in MODE 3 is deleted (addressed by new Condition N for when Required Action and associated CT not met).

TS 3.6.5, "Containment Spray and Cooling Systems"

- Required Action D.2
  - The option of calculating a RICT is applied to the action to restore all FCUs to OPERABLE status (for Condition D, One containment cooling FCU in each train inoperable).

TS 3.7.8, “Cooling Water (CL) System”

- Required Action note 3, which does not allow condition A (no safeguards CL pumps operable for one train) to exist greater than 7 days in any consecutive 30 day period, is deleted.

- Required Action A.1
    - The option of calculating a RICT is applied to the action to restore one safeguards CL pump to OPERABLE status (for Condition A).

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

The following individual Conditions, Required Actions and administrative controls identified below would be modified by the proposed change.

TS 3.3.1, “Reactor Trip System (RTS) Instrumentation”

- Condition O (previously N before insertion of new Conditions N) is revised to include a period at the end of the Condition, as identified in the mark-up TSs provided in Attachment 2 to the LAR.

- Required Action V.1 (previously S.1 before insertion of new Conditions N, P, and U)
    - The word “inoperable” is deleted (for Condition V, One trip mechanism inoperable for one Reactor Trip Breaker).

TS 3.7.2, “Main Steam Isolation Valves (MSIVs)”

- The applicability for TS 3.7.2 is modified by making “MODE 1” singular, instead of plural, “MODES 1,” so that it reads, “MODE 1, MODES 2 and 3 except when both MSIVs are closed.”

TS 3.7.8, “Cooling Water (CL) System”

- The vertical formatting alignment of required action note 1 is modified to be consistent with the alignment of required action notes 2 & 3.

TS 5.5.16, “Control Room Envelope Habitability Program

Underline is added to the TS 5.5.16 header, “Control Room Envelope Habitability Program (continued)” on TS page 5.0-31.

2.3 Regulatory Review

2.3.1 Applicable Regulations

Under Section 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 on 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.

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 50.36(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.

The regulations in 10 CFR 50.46 "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" require, in part, that nuclear power plants must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in the section.

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.

2.3.2 Commission Policy

The NRC provided details concerning the use of PRA in the "Final Policy Statement: Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," published in the

Federal Register (60 FR 42622; August 16, 1995). In this publication, the Commission wrote, in part:

> The Commission believes that an overall policy on the use of PRA methods in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach….
>
> PRA addresses a broad spectrum of initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for multiple and common cause failures. The treatment therefore goes beyond the single failure requirements in the deterministic approach. The probabilistic approach to regulation is, therefore, considered an extension and enhancement of traditional regulation by considering risk in a more coherent and complete manner….
>
> Therefore, the Commission believes that an overall policy on the use of PRA in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that promotes regulatory stability and efficiency. This policy statement sets forth the Commission's intention to encourage the use of PRA and to expand the scope of PRA applications in all nuclear regulatory matters to the extent supported by the state-of-the-art in terms of methods and data….
>
> Therefore, the Commission adopts the following policy statement regarding the expanded NRC use of PRA:
>
> (1) The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.
>
> (2) PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.
>
> (3) PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

(4) The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

2.3.3 Regulatory Guidance

Revision 3 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 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). The limits for ICCDP and ICLERP are consistent with the criteria for incremental core damage probability (ICDP) and incremental large early release probability (ILERP) from the Nuclear Management and Resources Council (NUMARC) 93-01, Revision 4A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," April 2011 (ADAMS Accession No. ML11116A198), guidance for managing the risk of on-line maintenance activities. The ICDP and ILERP are the limits on which licensee will base the RICT. This guidance was endorsed by the NRC staff in RG 1.160, Revision 3, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," May 2012 (ADAMS Accession No. ML113610098), for compliance with the Maintenance Rule, 10 CFR 50.65(a)(4). Tier 1 also addresses PRA acceptability, including the technical adequacy of the licensee's plant-specific PRA for the subject application. Cumulative risk of the proposed TS change is considered with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

- 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 3, 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 philosophy;

3. The proposed change maintains sufficient safety margins;

4. When proposed changes result in an increase in core damage frequency (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 LAR proposed adoption of TSTF-505, Revision 2, which 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 that exists at the time of the applicability of the Required Action and are updated when plant conditions change.

The 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 Completion Time (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 PRA methods and models provided in the LAR to determine if they are technically acceptable for use in the proposed RICTs. The NRC staff also reviewed the proposed RICT Program to determine if it provides the necessary administrative controls to permit completion time extensions.

3.1 <u>Review of Key Principles</u>

Regulatory Guide 1.174, Revision 3, and RG 1.177, Revision 1, 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 proposed use of RICTs against these key safety principles is discussed below.

3.1.1 Key Principle 1: Evaluation of Compliance with Current Regulations

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 5.5.18. 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 judgement. 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 3, and RG 1.177, Revision 1.

3.1.2 Key Principle 2: Evaluation of Defense-in-Depth

Defense-in-depth 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. Defense-in-depth 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 defense-in-depth 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 redundant and diverse key safety functions that provide 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 defense-in-depth will not be significantly impacted by such changes to the licensing basis. The LAR proposed 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 defense-in-depth 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. The increased use of compensatory measures potentially degrades defense-in-depth through increase reliance on programmatic activities as compensatory measures. The acceptability of potential degradation of defense-in-depth is discussed below for the applicable TS changes. 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.

3.1.2.1 Use of Compensatory Measures to Retain Defense-in-Depth

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. Topical Report 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.

Topical Report 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. Therefore, quantitative risk analysis, the qualitative considerations, and the prohibition on loss of

all trains of a required system assure a reasonable balance of defense-in-depth is maintained to ensure protection of public health and safety. The NRC staff evaluated defense-in-depth as described above for all TS conditions in the scope of the RICT program. A detailed evaluation of defense-in-depth for select LCO conditions is provided in Sections 3.1.2.2 through 3.1.2.6 below.

3.1.2.2 Evaluation of Electrical Power Systems

System Description

According to the Prairie Island Updated Safety Analysis Report (USAR), the source of power for plant loads during normal operation is each unit's main generator. The auxiliary electrical system, which includes multiple offsite AC sources, two emergency diesel generators (EDGs) per unit, and two 125 volt direct current (VDC) systems for each unit, provides reliable power to those auxiliaries required during any normal or emergency mode of plant operation, including plant start-up and shutdown. The design of the system is such that sufficient independence or isolation between the various sources of electrical power is provided in order to guard against concurrent loss of all auxiliary power.

Each main generator feeds electrical power at 20 kilovolts (kV) through its isolated phase bus to the associated generator step up transformer. The power requirements for station and unit auxiliaries are supplied by a main auxiliary transformer connected to the isolated phase bus, following practices that have been highly satisfactory for fossil-fueled and other nuclear units. Auxiliary power for startup, shutdown, and normal backup is supplied from Reserve Auxiliary Transformers designated 1R and 2RS. Transformer 1R is the normal backup source for Unit 1 and is connected to the 161 kV external power system. Transformer 2RS is the normal backup source for Unit 2 and is connected to the 345 kV external power system. Bus ties are provided to allow cross feeding should one transformer be out-of-service. Redundant offsite power sources are provided of sufficient capacity to supply all critical loads for either or both units. Each safeguards bus has a preferred and alternate offsite source consisting of a reserve auxiliary transformer and a cooling tower substation transformer, respectively. The cooling tower substation sources are not large enough to also serve as a redundant startup source.

To ensure continuity of supply for critical loads, emergency backup power is supplied from four onsite, quick-start EDGs. The EDGs are connected to the separate 4160 volt safeguards buses in each unit. Each EDG is started automatically on a safety injection (SI) signal from its unit or upon the occurrence of undervoltage on its corresponding 4160 volt bus. The EDG arrangement provides adequate capacity to supply the engineered safety features for the design basis accident in one unit, assuming the failure of a single active component in the system. The safeguards 4 kV buses are designed with cross-tie capability to the same train bus on the opposite unit (i.e., Bus 15 cross-tie to 25 and Bus 16 cross-tie to 26). The cross-tie consists of two 4 kV breakers in series with one breaker on each bus. The breakers must be closed by operators from the control room when desired and they will automatically open during the load shed cycle after an SI signal from the applicable unit. Each cross-tie breaker is open during normal operation and is typically only closed for short periods during testing.

The 125 VDC systems supply instrumentation, control, and motive power to safety related equipment. Redundant safety-related equipment is divided between the two DC subsystems associated with each unit such that loss of one DC subsystem does not affect redundant circuits. The safeguards 125 VDC electrical power system for each unit consists of two independent and redundant safety related DC electrical power subsystems (Train A and

Train B). The 125 VDC subsystems 11 and 12 serve Unit 1 and 125 VDC subsystems 21 and 22 serve Unit 2. Each subsystem consists of one 125 VDC battery, battery charger, and associated distribution equipment.

Technical Evaluation

The LAR requested to use the RICT Program to extend the existing CT for TS 3.8 and TS 3.3.4 conditions. The NRC staff's evaluation of the proposed changes considered a number of 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.

The NRC staff reviewed information pertaining to the proposed electrical power systems TS conditions in the application, the USAR, TS Bases, and applicable TS LCOs 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 (DSC) reflect the redundant or absolute minimum electrical power source/subsystem/component required to be operable by the LCOs to support the safety functions necessary to mitigate postulated design-basis accidents, safely shut down 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 TS condition satisfies its DSC. 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-related or non-safety-related) available at Prairie Island capable of performing the same safety function of the inoperable electrical power source/equipment. In addition, the NRC staff reviewed the proposed RMA examples in Enclosure 12 of the LAR and in the licensee's September 1, 2020, response to the NRC staff's request for additional information (RAI), dated July 7, 2020 (ADAMS Accession No. ML20192A144), for reasonable assurance that these RMAs are appropriate to monitor and control risk and to ensure adequate defense-in-depth for applicable TS conditions.

As discussed in Prairie Island USAR Section 1.2, Revision 36 (ADAMS Accession No. ML20118D368), "Prairie Island was designed and constructed to comply with NSPM's understanding of the intent of the Atomic Energy Commission (AEC) General Design Criteria (GDC) for Nuclear Power Plant Construction Permits (Appendix A to 10 CFR 50), as published on July 11, 1967." Specifically, Criterion 24, "Emergency Power for Protection Systems," states that in the event of all offsite power, sufficient alternate sources of power should be provided to permit the required functioning of the protection systems. Criterion 39, "Emergency Power for Engineered Safety Features," states that alternate power systems should 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 onsite power system and the offsite power system should each independently, provide this capacity assuming a failure of single active component in each power system.

The onsite safeguards AC distribution system is divided into redundant trains so that the loss of any one train does not prevent the minimum safety functions from being performed. Each train has two connections to the offsite power sources, and one to an onsite EDG. Offsite power is supplied to the unit switchyard(s) from the transmission network by five transmission lines.

From the switchyard(s), electrically and physically separated paths provide AC power, through step down station auxiliary transformers, to the 4 kV safeguards buses.

During the RICT Program entry for the proposed electrical TS conditions, when the LCO is not met due to the inoperable electrical power source or equipment, the redundancy required by the TS LCO (in operating modes), as specified by AEC GDC 24 and 39, will not be maintained. Therefore, the NRC staff finds that the design requirements are not temporarily met during the RICT Program entry for the proposed electrical power systems TS conditions since the redundancy required by the GDC 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 the GDC and LCO is not maintained, is allowed by 10 CFR 50.36(c)(2), which states, "When a limiting condition for operational 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 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. 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 Prairie Island TS 5.5.18. The RICT is limited to a maximum of 30 days and can only be used when there is no TS or PRA loss of function. 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.

The LAR stated that the proposed amendments would modify the TS requirements related to CTs for various required actions to provide the option to calculate a longer risk-informed CT. The LAR proposed to incorporate the RICT Program into its TSs for applicability only during Modes 1 and 2.

The NRC staff reviewed Attachment 4 of the LAR which identified the Prairie Island TS electrical power systems LCO conditions for TS 3.3.4 and TS section 3.8 (the changes are described in Section 2.2.3 of this SE) to be included as part of the RICT Program in accordance with TSTF-505, Revision 2. The NRC staff notes that Prairie Island TS 3.3.4 Condition C is the same as TS 3.8.1 Condition F, as found in the standard TS. The TSTF-505, Revision 2, model SE states that application of the RICT Program requires that there is no TS loss of function conditions. In addition, the NRC staff reviewed Table E1-1 provided in Enclosure 1 of the LAR for each of the above electrical power system TS LCO conditions to which the RICT Program is proposed to be applied. The NRC staff also reviewed information regarding the TSs such as proposed TS LCO condition, SSCs covered by TS LCO condition, SSCs modeled in PRA, function(s) covered by TS LCO condition, DSC, and PRA success criteria.

The NRC staff noted that in Table E1-1 of Enclosure 1 of the LAR, the licensee stated that the DSC for TS 3.8.1, Condition C – Two paths inoperable, are one qualified path to the grid for one safeguards bus. Therefore, the NRC staff requested the licensee clarify how each path/circuit

satisfies the DSC in terms of independence and capacity. The licensee's response stated the following:

> The design success criteria for TS 3.8.1 was incorrectly specified in Table E1-1 of Enclosure 1 of the LAR. The design success criteria for T.S. 3.8.1, conditions A, B, C, and D is:
>
>> One of two paths between the transmission grid and onsite 4kV Safeguards Distribution System or one of two diesel generators capable of supplying the onsite 4kV Safeguards Distribution System.
>
> Offsite power is supplied to the unit switchyard from the transmission network by five transmission lines. From the switchyard, electrically and physically separated paths provide AC power, through step down station auxiliary transformers, to the 4 kV safeguards buses. A path consists of all breakers, transformers, switches, cabling, and controls required to transmit power from the offsite transmission network to the safeguards buses. Each emergency diesel generator provides the third AC power source for each 4kV safeguards bus. The combination of multiple paths to the grid for each bus, emergency diesel generator for each bus, and two separate buses ensures the required independence and redundancy to ensure an available source of power.
>
> See Attachment 1 to this Enclosure for the revised Table E1-1 that reflects this correction and supersedes that provided in Enclosure 1 of the LAR.

The NRC staff verified that revised Table E1-1 clarified the DSC for TS 3.8.1, Conditions A, B, C, and D, as "One of two paths between the transmission grid and onsite 4 kV safeguards distribution system or one of two diesel generators capable of supplying the onsite 4 kV Safeguards Distribution System." The NRC staff finds that TS 3.8.1 Conditions A, B, C, and D do not involve a loss of safety function because either one remaining offsite circuit or one onsite power source is available to perform the design safety function, consistent with TSTF-505, Revision 2, and is, therefore, acceptable.

Table A4-1, "Cross-Reference of TSTF-505 and PINGP [Prairie Island Nuclear Generating Plant] Technical Specifications" of Attachment 4 of the LAR stated in the TS 3.3.4.C.5 comments section that the "PINGP TS 3.3.4 Required Action C.5 is equivalent to TS 3.8.1 Required Action F.1 in TSTF-505." The automatic load sequencer prevents a low voltage condition on the 4 kV safeguards bus that occurs when too much electrical load is added at once following EDG start-up. The result is that one 4 kV safeguards bus is not powered by the EDG, as designed. The condition is similar to having one diesel inoperable.

The NRC staff verified that Table E1-1 clarified the DSC for TS 3.3.4, Condition C as "One Automatic Load Sequencer per 4kV bus." The NRC staff found that TS 3.3.4, Condition C does not involve a loss of safety function because the opposite train's automatic load sequencer is available to perform the design safety function, consistent with TSTF-505, Revision 2, and is, therefore, acceptable.

The NRC staff noted that TS LCO conditions for direct current (DC) system 3.8.4 A through C, Inverter 3.8.7 A, and Distribution System 3.8.9 A through C included in the RICT Program do not cause a loss of safety function because the DSC requires one train or subsystem of AC or DC power sources are maintained, in accordance with Prairie Island design basis requirements.

Therefore, the staff concludes that the Prairie Island TS 3.8 for Electrical Power Systems to be included as part of the RICT Program is consistent with TSTF-505, Revision 2, and is, therefore, acceptable.

The LAR proposes a new TS requirement, 5.5.18, which requires that a RICT may not exceed 30 days. Therefore, up to 30 days would be permitted to restore the inoperable electrical power system train or channel equipment while the plant is in operational Modes 1 and 2. The NRC staff reviewed LAR Table E1-2, "In Scope TS/LCO Conditions RICT Estimate," and determined that electrical systems' RICT estimates do not exceed 30 days.

The NRC staff also reviewed LAR Enclosure 12 that describes the process for identification and implementation of RMAs applicable during extended CTs and provides examples of RMAs. Section 4 of Enclosure 12 provides the specific examples for unavailability of one diesel generator, one offsite source, and one battery charger.

The NRC staff notes that the Prairie Island protected equipment program identifies equipment that should be protected as part of compensatory measures and risk management actions to ensure that the minimum required equipment remains available to support plant operation. In addition, the actions specified above would provide additional assurance of the availability of the remaining equipment and adequate defense-in-depth. Therefore, the NRC staff finds that the examples of the RMAs associated with inoperable power sources provided are consistent with NEI 06-09-A.

NEI 06-09-A states the following regarding high-risk configurations:

> RMTS evaluations shall evaluate the instantaneous core damage frequency (CDF), instantaneous large early release frequency (LERF). If the SSC inoperability will be due to preplanned work, the configuration shall not be entered if the CDF is evaluated to be $\geq 10^{-3}$ events/year or the LERF is evaluated to be $\geq 10^{-4}$ events/year. If the SSC inoperability is due to an emergent event, if these limits are exceeded, the plant shall implement appropriate risk management actions to limit the extent and duration of the high risk configuration.

NEI 06-09-A, does not permit voluntary entry into a high-risk configuration but it allows entry in such configurations due to emergent events with implementation of appropriate risk management actions.

Table E1-2 of Enclosure 1 of the LAR provides RICT estimates for TS actions proposed to be in the scope of the RICT Program. However, RICT estimates for TS 3.8.4.B, TS 3.8.4.C, TS 3.8.9.A, and TS 3.8.9.B are not provided. In addition, Note #2 of Table E1-2 states:

> 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.

Since Note #2 of Table E1-2 appears to be inconsistent with NEI 06-09-A, which states that involuntary RICT entries into conditions of high instantaneous CDF or LERF would be also prohibited, RAI 20 requested clarification of the intent of this Note and the RMAs that would be implemented for these TS conditions. The licensee's response, dated September 1, 2020, stated the following:

> The requirements of NEI 06-09 will be followed regarding involuntary entries into high risk configurations.
>
> Note 2 was intended to explain that the calculated RICT estimate was conservative for the configuration and that less significant trains or sub-components within scope for the LCO could result in "inoperable" equipment but with a lower calculated risk, depending on what equipment was out of service.
>
> The RICT estimate for TS 3.8.4.B was based on the worst case of the four safeguards batteries out of service. The calculated risk varied significantly between the trains/units due to asymmetries in the PRA model caused by plant design. For example, the "A" train battery on both units exceeded the risk cap but the "B" train battery would have allowed a voluntary RICT entry. The worst case result was provided in the LAR.
>
> The RICT estimates for TS 3.8.4.C, TS 3.8.9.A, and TS 3.8.9.B were calculated based on taking the main 4kV bus or DC panel out of service. The TSs in question apply to multiple components within the AC and DC distribution systems. A bus, MCC [main control pane], or panel downstream of the main bus/panel would likely result in lower calculated risk because less equipment would be impacted. In this case, voluntary entry into a RICT may be possible depending on the magnitude of the calculated risk.
>
> . . .
>
> Risk-management actions implemented for TS 3.8.4.B, TS 3.8.4.C, TS 3.8.9.A, and TS 3.8.9.B would be dependent on the configuration at the time and which train/unit was impacted by the inoperable equipment. RMAs would include actions similar to those specified in LAR Enclosure 12 and would be very similar for all four LCOs due to the risk significance of the components involved.
>
> <u>TS 3.8.4.B (11 Battery inoperable example) RMAs</u>:
>
> - Perform a walkdown to validate standby/readiness condition of the "B" train ECCS components for Unit 1.
> - Perform a walkdown and validation of both 11 & 12 AFW trains to validate standby/readiness condition.
> - Perform a walkdown of and confirm availability of applicable suppression, detection and fire barriers for Fire Area 20 (Unit 1 4.16 kV Safeguards Switchgear Bus 16)
> - Notify the transmission system operator (TSO) of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred.

- Defer planned maintenance or testing.

TS 3.8.4.C and TS 3.8.9.B (DC Panel 11 inoperable example) RMAs

- Perform a walkdown to validate standby/readiness condition of the “B” train ECCS components for Unit 1.
- Perform a walkdown and validation of the 12 AFW train to validate standby/readiness condition.
- Perform a walkdown of and confirm availability of applicable suppression, detection and fire barriers for the following Fire Areas:
  - Fire Area 20 (Unit 1 4.16 kV Safeguards Switchgear Bus 16)
  - Fire Area 58 (Auxiliary Building Ground Floor)
- Notify the transmission system operator (TSO) of the configuration so that any planned activities with the potential to cause a grid disturbance are differed.
- Defer planned maintenance or testing

TS 3.8.9.A (Bus 15 inoperable example) RMAs

- Perform a walkdown to validate standby/readiness condition of the “B” train ECCS components for Unit 1.
- Perform a walkdown and validation of the 12 AFW train to validate standby/readiness condition.
- Perform a walkdown of and confirm availability of applicable suppression, detection and fire barriers for the following Fire Areas:
- Fire Area 20 (Unit 1 4.16 kV Safeguards Switchgear Bus 16)
- Fire Area 58 (Auxiliary Building Ground Floor)
- Notify the transmission system operator (TSO) of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred.
- Defer planned maintenance or testing.

Based on the NRC staff’s review of the information provided in the LAR, the staff determined that the compensatory measures or RMAs for maintaining operability of the remaining train or channel(s) are reasonable and consistent with the guidance provided in NEI 06-09-A and TSTF-505, Revision 2. The NRC staff also determined that at least one operable train (i.e., subsystem) is available to support the safety function(s) of an onsite electrical power system and offsite electric power subsystem to permit functioning of SSCs important to safety.

Evaluation of Electrical Power Systems Conclusion

The NRC staff reviewed the proposed TS changes and supporting documentation. Based on the evaluations above, the 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 (a) the capability of the systems to perform their safety functions (assuming no additional failures) is maintained, and (b) 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.

Therefore, the NRC staff concludes that the proposed changes are acceptable and consistent with the principle of defense-in-depth.

3.1.2.3 Evaluation of Instrumentation and Control Systems

The LAR proposed to use the RICT Program to extend the existing CT for the following instrumentation and control (I&C) systems TS actions. The NRC staff's evaluation of the proposed changes considered a number of potential plant conditions allowed by the proposed TSs and considered what redundant or diverse means were available to assist in responding to various plant conditions.

The plant conditions evaluated are discussed in more detail below.

TS 3.3.1	*Reactor Trip System (RTS) Instrumentation*
LCO:	*The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.*

Current:

Condition B:	One Manual Reactor Trip channel inoperable.
Required Action:	B.1 Restore channel to OPERABLE status.
Completion Time:	48 hours
Required Action:	B.2 Be in MODE 3
Completion Time:	54 hours

Proposed:

Condition B:	One Manual Reactor Trip channel inoperable.
Required Action:	B.1 Restore channel to OPERABLE status.
Completion Time:	48 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

Condition B applies to the manual reactor trip in Mode 1 or 2. With one manual reactor trip channel inoperable, current Condition B allows 48 hours to restore channel to operable status. For this condition, the RICT insert "or in accordance with the Risk Informed Completion Time Program" has been added below the CT for Required Action B.1.

Required Action B.2 to be in Mode 3 in CT of 54 hours has been deleted. However, insert TS 3.3.1, Condition W (addressed below), has been added to "Be in MODE 3" in 6 hours when the Required Action and associated CT of Condition B is not met.

One-out-of-two channels are required to trip. With one channel inoperable, one-out-of-one channel will be available to achieve the required trip function even though the redundancy is lost.

Current:

Condition D:	One Power Range Neutron Flux channel inoperable.
Required Action:	D.1.1 Place channel in trip.
Completion Time:	6 hours
Required Action:	D.1.2 Perform SR 3.2.4.2.
Completion Time:	Once per12 hours
Required Action:	D.2 Be in MODE 3
Completion Time:	12 hours

Proposed:

Condition D:	One Power Range Neutron Flux channel inoperable.
Required Action:	D.1.1 Place channel in trip.
Completion Time:	6 hours OR In accordance with the Risk Informed Completion Time Program
Required Action:	D.1.2 Perform SR 3.2.4.2.
Completion Time:	Once per 12 hours

Evaluation

With one power range neutron flux channel inoperable, Condition D currently allows 6 hours to restore the channel to operable status or to place it in the tripped condition and it allows 12 hours to perform SR 3.2.4.2. Condition D applies to the following reactor trip Functions:

- Power Range Neutron Flux-High;
- Power Range Neutron Flux-Low;
- Power Range Neutron Flux-High Positive Rate;
- Power Range Neutron Flux-High Negative Rate.

For this condition, the RICT insert “or in accordance with the Risk Informed Completion Time Program” has been added between required CT for Required Action D.1.1 and required CT for Required Action D.1.2.

Required Action D.2 to be in Mode 3 in CT of 12 hours has been deleted. However, insert TS 3.3.1, Condition W (addressed below), has been added to “Be in MODE 3” in 6 hours when the Required Action and associated CT of Condition D is not met.

Two out of the four channels are required to trip. With one channel inoperable, two out of remaining three channels will be available for trip to achieve the required trip function even though the redundancy has decreased.

Current:

Condition E:	One channel inoperable.
Required Action:	E.1 Place channel in trip.
Completion Time:	6 hours
Required Action:	E.2 Be in MODE 3
Completion Time:	12 hours

Proposed:

Condition E: One channel inoperable.
Required Action: E.1 Place channel in trip.
Completion Time: 6 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

With one channel inoperable, Condition E currently allows 6 hours to restore the channel to operable status or to place it in the tripped condition, and applies to the following reactor trip functions:

- Overtemperature ΔT
- Overpower ΔT
- Pressurizer Pressure-High
- SG Water Level-Low Low

For this condition, the RICT insert “or in accordance with the Risk Informed Completion Time Program” has been added below the CT for Required Action E.1.

Required Action E.2 to be in Mode 3 in CT of 12 hours has been deleted. However, insert TS 3.3.1, Condition W (addressed below), has been added to “Be in MODE 3” in 6 hours when the Required Action and associated CT of Condition E is not met.

Two-out-of-three or two-out-of-four channels are required to trip. With one channel inoperable, two-out-of-two or two-out-of-three channels will be available to achieve the required trip function even though the redundancy is decreased or lost (pressurizer pressure-high, SG water level-low low.)

Current:

Condition K: One channel inoperable.
Required Action: K.1 Place channel in trip.
Completion Time: 6 hours
Required Action: K.2 Reduce THERMAL POWER to < P-7 and P-8.
Completion Time: 12 hours

Proposed:

Condition K: One channel inoperable.
Required Action: K.1 Place channel in trip.
Completion Time: 6 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

With one channel inoperable, Condition K currently allows 6 hours to restore the channel to operable status or to place it in the tripped condition and applies to the following reactor trip functions:

- Pressurizer Pressure-Low
- Pressurizer Water Level-High
- Reactor Coolant Flow-Low (single loop)
- Reactor Coolant Flow-Low (both loops)

Required Action K.1 states "Place channel in trip" at the end of 6 hours CT. For this function, the RICT insert "or in accordance with the Risk Informed Completion Time Program" has been added below the CT for Required Action K.1.

Required Action K.2 to reduce thermal power to < P-7 and P-8 in 12 hours has been deleted. However, Condition N (addressed below) has been added to allow 6 hours to reduce the thermal power to below < P-7 and P-8 if Condition K is not met.

Two-out-of-three or two-out-of-four channels are required to trip. With one channel inoperable, two-out-of-two remaining channels or two-out-of-three remaining channels will be available to achieve the required trip function even though the redundancy is decreased or lost (pressurizer water level-high, reactor coolant flow-low (single loop), reactor coolant flow-low (both loops).)

Current:

Condition L:	One or both channel(s) inoperable on one bus.
Required Action:	L.1 Place channel(s) in trip.
Completion Time:	6 hours
Required Action:	L.2 Reduce THERMAL POWER to < P-7 and P-8.
Completion Time:	12 hours

Proposed:

Condition L:	One or both channel(s) inoperable on one bus.
Required Action:	L.1 Place channel(s) in trip.
Completion Time:	6 hours OR In accordance with the Risk Informed Completion Time Program The application of the Risk Informed Completion Time Program is not applicable when more than one channel on one bus is inoperable

Evaluation

Condition L applies to the Loss of Reactor Coolant Pump Underfrequency 4 kV Buses (11 and 12 for Unit 1 and 21 and 22 for Unit 2) and Undervoltage on 4 kV Buses (11 and 12 for Unit 1 and 21 and 22 for Unit 2). With one or both channel(s) inoperable on one bus, Condition L currently allows 6 hours to restore the channel(s) to operable status or to place the channel(s) in the tripped condition. For this condition, the RICT insert "or in accordance with the Risk Informed Completion Time Program" has been added below the CT for Required Action L.1. Also, a RICT NOTE has been added stating that the application of the Risk Informed

Completion Time Program is not applicable when more than one channel on one bus is inoperable.

Required Action L.2 to reduce thermal power to < P-7 and P-8 in 12 hours has been deleted. However, Condition N (addressed below) has been added to allow 6 hours to reduce the thermal power to below < P-7 and P-8 if Condition L is not met.

Condition L provides protection against violating the departure from nucleate boiling (DNB) ratio limit due to a loss of flow in both RCS loops from a major network frequency disturbance. A loss of frequency detected on both reactor coolant pump (RCP) buses will initiate a trip of both RCP breakers. As stated above, due to the added RICT NOTE to Condition L the RICT Program is only applicable when one channel on one bus is inoperable. With one channel inoperable, the remaining channels will be available to achieve the required trip function even though the redundancy is decreased or lost.

Current:

Condition M:	One Reactor Coolant Pump Breaker Open channel inoperable.
Required Action:	M.1 Restore channel to OPERABLE status.
Completion Time:	48 hours
Required Action:	M.2 Reduce THERMAL POWER to < P-7 and P-8.
Completion Time:	54 hours

Proposed:

Condition M:	One Reactor Coolant Pump Breaker Open channel inoperable.
Required Action:	M.1 Restore channel to OPERABLE status.
Completion Time:	48 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

Condition M applies to the RCP breaker open reactor trip Function. With one reactor coolant pump breaker open channel inoperable, Condition M currently allow 48 hours to restore the channel to operable status. For this condition, the RICT insert “or in accordance with the Risk Informed Completion Time Program” has been added below the CT for Required Action M.1. Also, a RICT NOTE has been added stating that the application of the Risk Informed Completion Time Program is not applicable when thermal power is below P-8 and above P-7.

Required Action M.2 to reduce thermal power to < P-7 and P-8 in 54 hours has been deleted. However, Condition N (addressed below) has been added to allow 6 hours to reduce the thermal power to below < P-7 and P-8 if Condition M is not met.

In Mode 1 above the P-7 or P-8 setpoints, a loss of flow in an RCS loop could result in DNB conditions in the core. Below the P-7 and P-8 setpoints, all reactor trips on low flow are automatically blocked since there is insufficient heat production to generate DNB conditions. If one RCP breaker is open above the P-8 setpoint, a reactor trip is initiated. If both RCP breakers are open above the P-7 setpoint, a reactor trip is initiated.

The licensee stated in its September 1, 2020, letter that “This condition is a loss of function when thermal power is greater than the P-7 interlock and less than the P-8 interlock. Therefore,

the TS markup has been updated to add a note to Condition M excluding the use of RICT in this condition." Based on the addition of the note, the RICT is not applicable during a loss of function, when thermal power is below P-8 and above P-7. Therefore, when there is not a loss of function (i.e., above P8), the remaining channels will be available to achieve the required trip function even though the redundancy is lost.

Add new Condition N:

If the Required Action and associated CT of Condition K, L, or M is not met, thermal power must be reduced below the P-7 and P-8 setpoints within the next 6 hours.

Evaluation

This places the unit at a power level where the LCO is no longer applicable. The CT of 6 hours is reasonable, based on operating experience, to reach the applicable power level from full power in an orderly manner and without challenging unit systems.

Current:

Condition N:	One Turbine Trip channel inoperable
Required Action:	N.1 Place channel in trip.
Completion Time:	6 hours
Required Action:	N.2 Reduce THERMAL POWER to < P-9
Completion Time:	12 hours

Proposed:

Condition O:	One Turbine Trip channel inoperable
Required Action:	O.1 Place channel in trip.
Completion Time:	6 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

Condition O (previously Condition N) applies to turbine trip on low autostop oil pressure or on turbine stop valve closure. Required Action N.1 has been re-labelled as O.1 and RICT insert, "or in accordance with the Risk Informed Completion Time Program" has been added below the CT for Required Action O.1.

Previous Required Action N.2 to reduce thermal power to < P-9 in 12 hours has been deleted and a new Condition P (addressed below) has been added. The new Condition P states, "Required Action and associated CT of Condition O not met." The associated Required Action is to reduce thermal power to <P-9 with CT of 6 hours.

Two-out-of-three channels are required to trip for turbine trip on low autostop oil pressure. With one channel inoperable, two-out-of-two remaining channels will be available to achieve the required trip function even though the redundancy is lost.

In the enclosure to its September 1, 2020, letter, the licensee stated that "either the Turbine Stop Valve Closure channels OR the Low Autostop Oil pressure channels are sufficient to cause the reactor trip based on turbine trip." With one turbine stop valve closure channel

inoperable, the low autostop oil pressure channels will be available to achieve the required trip function even though the redundancy is lost.

Add new Condition P:

If the Required Action and associated CT of Condition O is not met, thermal power must be reduced below the P-9 setpoint within 6 hours.

Evaluation

This places the unit at a power level where the LCO is no longer applicable. The Completion Time of 6 hours is reasonable, based on operating experience, to reach the applicable power level from full power in an orderly manner and without challenging unit systems.
Current:

Condition O:	One train inoperable.
Required Action:	O.1 Restore train to OPERABLE status.
Completion Time:	6 hours
Required Action:	O.2 Be in MODE 3
Completion Time:	12 hours

Proposed:

Condition Q:	One train inoperable.
Required Action:	Q.1 Restore train to OPERABLE status.
Completion Time:	6 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

Condition Q (previously Condition O) applies to the safety injection (SI) Input from ESFAS reactor trip and the RTS automatic trip logic in Modes 1 and 2. With one train inoperable, it is required to be operable within the allowed time of 6 hours to restore the train to OPERABLE status. Previous Required Action O.1 has been re-labelled as Q.1 and the RICT insert has been added below the CT for the new Required Action Q.1.

Previous Required Action O.2 to be in Mode 3 in CT of 12 hours has been deleted. However, insert TS 3.3.1, Condition W (addressed below), has been added to “Be in MODE 3” in 6 hours when the required action and associated CT of Condition Q is not met.

One-out-of-two trains are required to achieve the safety function. With one train inoperable, the second train will be available for initiation to achieve the required function even though the redundancy is lost when only one train is available.

Current:

Condition P:	One RTB [reactor trip breaker] train inoperable.
Required Action:	P.1 Restore train to OPERABLE status.
Completion Time:	1 hour
Required Action:	P.2 Be in MODE 3
Completion Time:	7 hours

Proposed:

Condition R:	One RTB train inoperable.
Required Action:	R.1 Restore train to OPERABLE status.
Completion Time:	1 hour OR In accordance with the Risk Informed Completion Time Program

Evaluation

Condition R (previously Condition P) applies to the RTBs in Modes 1 and 2. Previous Required Action P.1 has been re-labelled as R.1 and the RICT insert, "or in accordance with the Risk Informed Completion Time Program" has been added below the CT for the new Required Action R.1.

Previous Required Action P.2 to be in Mode 3 in CT of 7 hours has been deleted. However, insert TS 3.3.1, Condition W (addressed below), has been added to "Be in MODE 3" in 6 hours when the required action and associated CT of Condition R is not met.

One-out-of-two trains are required to achieve the safety function. With one train inoperable, the second train will be available for initiation to achieve the required function even though the redundancy is lost when only one train is available.

Current:

Condition Q:	One or more channels inoperable.
Required Action:	Q.1 Verify interlock is in required state for existing unit conditions.
Completion Time:	1 hour
Required Action:	Q.2 Be in MODE 3
Completion Time:	7 hours

Proposed:

Condition S:	One or more channels inoperable.
Required Action:	S.1 Verify interlock is in required state for existing unit conditions.
Completion Time:	1 hour

Evaluation

Condition S (previously Condition Q) applies to interlocks P-6 (Intermediate range neutron flux) and P-10 (power range neutron flux) when one or more channels are inoperable in one-out-of-two or two-out-of-four coincident logic. The associated interlocks must be verified to be placed in the appropriate position within 1 hour. Previous Required Action Q.1 has been re-labelled as S.1.

Previous Required Action Q.2 to be in Mode 3 in CT of 7 hours has been deleted. However, insert TS 3.3.1 Condition W (addressed below), has been added to “Be in MODE 3” in 6 hours when the Required Action and associated CT of Condition S is not met.

The changes do not apply a RICT and do not alter the required actions or the allowed outage times. Therefore, the changes are acceptable to the NRC staff.

Current:

Condition R:	One or more channels inoperable.
Required Action:	R.1 Verify interlock is in required state for existing unit conditions.
Completion Time:	1 hour
Required Action:	R.2 Be in MODE 3
Completion Time:	7 hours

Proposed:

Condition T:	One or more channels inoperable.
Required Action:	T.1 Verify interlock is in required state for existing unit conditions.
Completion Time:	1 hour

Evaluation

Condition T (previously Condition R) applies to interlocks P-7 (low power reactor trips block), P-8 (power range neutron flux), and P-9 (power range neutron flux) when one or more channels are inoperable in one-out-of-two or two-out-of-four coincident logic. The interlocks must be confirmed to be placed in the appropriate position within the CT of 1 hour. Previous Required Action R.1 has been re-labelled as T.1.

Previous Required Action R.2 to be in Mode 3 in CT of 7 hours has been deleted. However, insert TS 3.3.1 Condition U (addressed below) has been added. The new Condition U states “Required Action and associated Completion Time of Condition T not met.” The associated Required Action is to be in Mode 2 in 6 hours.

The changes do not apply a RICT and do not alter the required actions or the allowed outage times. Therefore, the changes are acceptable to the NRC staff.

Add new Condition U:

If the Required Action and associated Completion Time of Condition T is not met, the unit must be placed in MODE 2 within 6 hours.

Evaluation

The CT of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

Current:

Condition S:	One trip mechanism inoperable for one RTB.
Required Action:	S.1 Restore inoperable trip mechanism to OPERABLE status.
Completion Time:	48 hours
Required Action:	S.2 Be in MODE 3
Completion Time:	54 hours

Proposed:

Condition V:	One trip mechanism inoperable for one RTB.
Required Action:	V.1 Restore inoperable trip mechanism to OPERABLE status.
Completion Time:	48 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

Condition V (previously Condition S) applies to the RTB undervoltage and shunt trip mechanisms, or diverse trip features, in Modes 1 and 2. With one trip mechanism inoperable for one RTB, restore trip mechanism to OPERABLE status within 48 hours. Previous Required Action S.1 has been re-labelled as V.1 and the RICT insert “or in accordance with the Risk Informed Completion Time Program” has been added below the CT for the new Required Action V.1.

Previous Required Action S.2 to be in Mode 3 in CT of 54 hours has been deleted. However, insert TS 3.3.1, Condition W (addressed below), has been added to “Be in MODE 3” in 6 hours when the required action and associated CT of Condition R is not met.

The LCO requires both the undervoltage and shunt trip mechanisms to be OPERABLE for each RTB that is in service. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal. With one trip mechanism inoperable for one RTB, it can still perform its required function, while no single trip mechanism failure will prevent opening of the other RTB on a valid signal and achieve the required function even though the redundancy is decreased or lost.

One-out-of-two trains are required to achieve the safety function. With one train inoperable, the second train will be available for initiation to achieve the required function even though the redundancy is lost when only one train is available.

Add new Condition W:

If the Required Action and associated Completion Time of Condition B, D, E, Q, R, S, or V is not met, the unit must be placed in MODE 3 within 6 hours. With the unit in MODE 3, ACTION K would apply to any inoperable RTB, RTB trip mechanism, or to any inoperable Manual Reactor Trip Function if the Rod Control System is capable of rod withdrawal or one or more rods are not fully inserted.

Evaluation

The CT of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

TS 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation
LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

Current:

Condition B:	One channel or train inoperable.
Required Action:	B.1 Restore channel or train to OPERABLE status.
Completion Time:	48 hours
Required Action:	B.2.1 Be in MODE 3
Completion Time:	54 hours
Required Action:	B.2.2 Be in MODE 5
Completion Time:	84 hours

Proposed:

Condition B:	One channel or train inoperable.
Required Action:	B.1 Restore channel or train to OPERABLE status.
Completion Time:	48 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

With one channel or train inoperable, Condition B currently allows 48 hours to restore the channel or train to operable status and applies to the manual initiation of:

- Safety Injection (SI)
- Containment Spray (CS)
- Containment Isolation (CI)

For this condition, the RICT insert “or in accordance with the Risk Informed Completion Time Program” has been added below the CT for Required Action B.1. Also, a RICT NOTE has been added stating that the application of the Risk Informed Completion Time Program is not applicable to Function 2.a.

Required Actions B.2.1 and B.2.2 to be in Mode 3 in CT of 54 hours and to be in Mode 5 in CT of 84 hours, respectively, have been deleted. However, insert TS 3.3.2, Condition L (addressed below), has been added to “Be in MODE 3” in 6 hours and to be in Mode 5 in 36 hours, when the Required Action and associated CT of Condition B is not met.

One-out-of-two channels are required to trip. With one channel inoperable, one-out-of-one channel will be available to achieve the required trip function even though the redundancy is lost.

Current:

Condition C:	One train inoperable.
Required Action:	C.1 Restore train to OPERABLE status.
Completion Time:	6 hours
Required Action:	C.2.1 Be in MODE 3
Completion Time:	12 hours
Required Action:	C.2.2 Be in MODE 5
Completion Time:	42 hours

Proposed:

Condition C:	One train inoperable.
Required Action:	C.1 Restore train to OPERABLE status.
Completion Time:	6 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

With one actuation train inoperable, Condition C currently allows 6 hours to restore inoperable train to operable status and applies to the automatic actuation relay logic for the following functions:

- SI
- CS
- CI

For this condition, the RICT insert “or in accordance with the Risk Informed Completion Time Program” has been added below the CT for Required Action C.1.

Required Actions C.2.1 and C.2.2 to be in Mode 3 in CT of 12 hours and to be in Mode 5 in CT of 42 hours, respectively, have been deleted. However, insert TS 3.3.2, Condition L (addressed below), has been added to be in Mode 3 in 6 hours and Mode 5 in 36 hours, when the Required Action and associated CT of Condition C is not met.

One-out-of-two trains are required to achieve the safety function. With one train inoperable, the second train will be available for initiation to achieve the required function even though the redundancy is lost when only one train is available.

Current:

Condition D:	One channel inoperable.
Required Action:	D.1 Place channel in trip.
Completion Time:	6 hours
Required Action:	D.2.1 Be in MODE 3
Completion Time:	12 hours
Required Action:	D.2.2 Be in MODE 4
Completion Time:	18 hours

Proposed:

Condition D:	One channel inoperable.
Required Action:	D.1 Place channel in trip.
Completion Time:	6 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

With one channel inoperable, Condition D currently allows 6 hours to restore the channel to operable status or to place it in the tripped condition, and applies to:

- Safety Injection High Containment Pressure
- Safety Injection Pressurizer Low Pressure
- Safety Injection Steam Line Low Pressure
- Steam Line Isolation High High Containment Pressure
- Steam Line Isolation High Steam Flow Coincident with Safety Injection and Coincident with Low Low Tavg
- Steam Line Isolation High Steam Flow Coincident with Safety Injection
- Auxiliary Feedwater Low Low SG Water Level

For this condition, the RICT insert “or in accordance with the Risk Informed Completion Time Program” has been added below the CT for Required Action D.1.

Required Actions D.2.1 and D.2.2 to be in Mode 3 in CT of 12 hours and to be in Mode 4 in CT of 18 hours, respectively, have been deleted. However, insert TS 3.3.2, Condition M (addressed below), has been added to be in MODE 3 in 6 hours and MODE 4 in 12 hours, when the Required Action and associated CT of Condition D is not met.

Two-out-of-three channels are required to achieve the safety function. With one channel inoperable, the remaining channels will be available for initiation to achieve the required function even though the redundancy has decreased.

Current:

Condition E:	One or more Containment Pressure channel(s) inoperable.
Required Action:	E.1.1 Place inoperable channel(s) in trip.
Completion Time:	6 hours
Required Action:	E.1.2 Verify one channel per pair OPERABLE.
Completion Time:	6 hours
Required Action:	E.2.1 Be in MODE 3
Completion Time:	12 hours
Required Action:	E.2.2 Be in MODE 4
Completion Time:	18 hours

Proposed:

Condition E:	One or more Containment Pressure channel(s) inoperable.
Required Action:	E.1.1 Place inoperable channel(s) in trip.
Completion Time:	6 hours OR In accordance with the Risk Informed Completion Time Program
Required Action:	E.1.2 Verify one channel per pair OPERABLE.
Completion Time:	6 hours

Evaluation

With one channel inoperable, Condition E currently allows 6 hours to restore the channel to operable status and verify one channel per pair is operable, or to place it in the tripped condition. Condition E applies to CS high high containment pressure. For this condition, the RICT insert "or in accordance with the Risk Informed Completion Time Program" has been added between required CT for action E.1.1 and required CT for action E.1.2.

Required Actions E.2.1 and E.2.2 to be in Mode 3 in CT of 12 hours and to be in Mode 4 in CT of 18 hours, respectively, have been deleted. However, insert TS 3.3.2, Condition M (addressed below), has been added to be in MODE 3 in 6 hours and MODE 4 in 12 hours, when the Required Action and associated CT of Condition E is not met.

CS high high containment pressure signal provides protection against a loss-of-coolant accident (LOCA) or a steam line break inside containment. It uses three sets of two channels, each set combined in a one-out-of-two configuration, with these outputs combined so that three sets tripped initiates CS. Condition E only allows the risk informing of conditions where one of the two is inoperable. With one channel inoperable, one-out-of-one channel will be available to achieve the required trip function in each set, even though the redundancy is lost.

Current:

Condition F:	One channel or train inoperable.
Required Action:	F.1 Restore channel or train to OPERABLE status.
Completion Time:	48 hours
Required Action:	F.2.1 Be in MODE 3
Completion Time:	54 hours
Required Action:	F.2.2 Be in MODE 4
Completion Time:	60 hours

Proposed:

Condition F:	One channel or train inoperable.
Required Action:	F.1 Restore channel or train to OPERABLE status.
Completion Time:	48 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

With one channel or train inoperable, Condition F currently allows 48 hours to restore the channel or train to operable status and applies to the manual initiation of steam line isolation.

For this condition, the RICT insert “or in accordance with the Risk Informed Completion Time Program” has been added below the CT for Required Action F.1.

Required Actions F.2.1 and F.2.2 to be in Mode 3 in CT of 54 hours and to be in Mode 4 in CT of 60 hours, respectively, have been deleted. However, insert TS 3.3.2, Condition M (addressed below), has been added to be in Mode 3 in 6 hours and Mode 4 in 12 hours, when the Required Action and associated CT of Condition F is not met.

Isolation of the main steam lines provides protection in the event of a steam line break inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG at most. Manual initiation of steam line isolation can be accomplished from the control room. There are two switches in the control room, one for each MSIV. The LCO requires one channel per loop to be OPERABLE. With one channel or train inoperable, the other channels will be available to achieve the required function, even though the redundancy is lost.

Current:

Condition G:	One train inoperable.
Required Action:	G.1 Restore train to OPERABLE status.
Completion Time:	6 hours
Required Action:	G.2.1 Be in MODE 3
Completion Time:	12 hours
Required Action:	G.2.2 Be in MODE 4
Completion Time:	18 hours

Proposed:

Condition G:	One train inoperable.
Required Action:	G.1 Restore train to OPERABLE status.
Completion Time:	6 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

With one train inoperable, Condition G currently allows 6 hours to restore channel to operable status and applies to the automatic actuation relay logic for the following functions:

- Steam Line Isolation
- Feedwater Isolation

For this condition, the RICT insert “or in accordance with the Risk Informed Completion Time Program” has been added below the CT for Required Action G.1.

Required Actions G.2.1 and G.2.2 to be in Mode 3 in CT of 12 hours and to be in Mode 4 in CT of 18 hours, respectively, have been deleted. However, insert TS 3.3.2, Condition M (addressed below), has been added to be in Mode 3 in 6 hours and Mode 4 in 12 hours, when the Required Action and associated CT of Condition G is not met.

One-out-of-two trains are required to achieve the safety function. With one train inoperable, the second train will be available for initiation to achieve the required function even though the redundancy is lost when only one train is available.

Current:

Condition H:	One channel inoperable.
Required Action:	H.1 Place channel in trip.
Completion Time:	6 hours
Required Action:	H.2. Be in MODE 3
Completion Time:	12 hours

Proposed:

Condition H:	One channel inoperable.
Required Action:	H.1 Place channel in trip.
Completion Time:	6 hours OR In accordance with the Risk Informed Completion Time Program

Evaluation

With one channel inoperable, Condition H currently allows 6 hours to restore the channel to operable status and applies to feedwater isolation high high SG water level. For this condition, the RICT insert “or in accordance with the Risk Informed Completion Time Program” has been added below the CT for Required Action H.1.

Required Action H.2 to be in Mode 3 in CT of 12 hours has been deleted. However, insert TS 3.3.2, Condition N (addressed below), has been added to “Be in MODE 3” in 6 hours when the Required Action and associated CT of Condition H is not met.

Two out of the three channels are required to trip. With one channel inoperable, two out of remaining two channels will be available for trip to achieve the required function even though the redundancy is lost.

Current:

Condition I:	One or both channel(s) inoperable on one bus.
Required Action:	I.1 Place channel(s) in trip.
Completion Time:	6 hours
Required Action:	I.2. Be in MODE 3
Completion Time:	12 hours

Proposed:

Condition I:	One or both channel(s) inoperable on one bus.
Required Action:	I.1 Place channel(s) in trip.
Completion Time:	6 hours OR In accordance with the Risk Informed Completion Time Program The application of the Risk Informed Completion Time Program is not applicable when more than one channel on one bus is inoperable

Evaluation

Condition I applies to auxiliary feedwater undervoltage on 4kV Buses (11 and 12 for Unit 1 and 21 and 22 for Unit 2). With one or both channel(s) inoperable on one bus, Condition I currently allows 6 hours to restore the channel(s) to operable status or to place the channel(s) in the tripped condition. For this condition, the RICT insert “or in accordance with the Risk Informed Completion Time Program” has been added below the CT for Required Action I.1. Also, a RICT NOTE has been added stating that the application of the Risk Informed Completion Time Program is not applicable when more than one channel on one bus is inoperable.

Required Action I.2 to be in Mode 3 in CT of 12 hours has been deleted. However, insert TS 3.3.2, Condition N (addressed below), has been added to “Be in MODE 3” in 6 hours when the Required Action and associated CT of Condition I is not met.

Once one or both of the inoperable channel(s) is placed in trip, the Function is then in a partial trip condition where one-out-of-two channels on the other bus will result in actuation. As stated above, due to the added RICT NOTE to Condition I, the RICT Program is only applicable when one channel on one bus is inoperable. With only one channel on one bus inoperable, the remaining channels and buses will be available to achieve the required trip function even though the redundancy is decreased or lost.

Add new Condition L:

If the Required Action and associated Completion Time of Condition B or C is not met, the unit must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the unit in MODE 3 within 6 hours and MODE 5 within 36 hours.

Evaluation

The allowed CTs are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

Add new Condition M:

If the Required Action and associated Completion Time of Condition D, E, F, or G is not met, the unit must be placed in MODE 3 within 6 hours and MODE 4 within 12 hours.

Evaluation

The allowed CTs are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

Add new Condition N:

If the Required Action and associated Completion Time of Condition H or I is not met, the unit must be placed in MODE 3 within 6 hours.

Evaluation

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, these Functions are no longer required OPERABLE. \

Defense-in-Depth for I&C TS Functions

Enclosure 1 of the LAR provided information about diverse means of achieving the required function. In RAI 18, the NRC staff requested additional information on the defense-in-depth for I&C TS functions. The licensee's response explained how the amendment request meets the defense-in-depth guidance contained in RG 1.174, Section 2.1.1 for every affected USAR Chapter 14 design basis accident. Based on the information provided, the NRC staff finds that at least one redundant or diverse means is available and the defense-in-depth guidance in RG 1.174 is met.

Conclusion of Evaluation of I&C Systems

Since the LAR 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 systems 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, SI, or containment isolation) remain available during the use of the RICT. The NRC staff reviewed the 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 defense-in-depth to accomplish the safety functions, allowing for the extension of CTs in accordance with the RICT Program. Therefore, the NRC staff finds that the 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 I&C redundancy is reduced, the CT extensions implemented in accordance with the RICT Program are acceptable because: (a) the capability of the I&C systems to perform their safety functions is maintained, (b) redundant or diverse means to accomplish the safety functions exist, and (c) the licensee will identify and implement RMAs to monitor and control risk in accordance with the RICT Program.

3.1.2.4 Evaluation of ECCS

The LAR proposes to modify the ECCS TS requirements in TS sections 3.5, 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 while maintaining the capability of the system to perform its safety function, is consistent with defense-in-depth principles, as discussed below.

The LAR proposed to modify the ECCS TS requirements to permit extending the selected CTs using the RICT Program, in accordance with NEI 06-09-A. The licensee's ECCS is a two-train, fully redundant engineered safety feature (ESF). The ECCS has been designed and proven by

the Prairie Island USAR Chapter 14 (ADAMS Accession No. ML20118D375) LOCA analysis to withstand any single credible active failure during injection, or an active/passive failure during recirculation and maintain the performance objectives desired in the Prairie Island USAR Subsection 6.2.1.1. With one ECCS train inoperable, 100 percent of the ECCS flow is provided by the remaining OPERABLE ECCS train. The ECCS train is designed to accept a single failure following its initiation without loss of its protective function. The ECCS design will allow for the failure of any single active component in the ECCS itself, or in the essential associated service systems at any time during the period required for ECCS operation following the initiating event.

A single passive failure analysis is presented in the Prairie Island USAR Table 6.2-8a. The analysis demonstrates that the ECCS can sustain a single passive failure during the long-term cooling phase, and still preserve an intact flow path to the reactor core to supply enough flow to keep the fuel assemblies covered and affect the removal of decay heat, which satisfies the 10 CFR 50.46 "Acceptance Criteria for Emergency Core Cooling System for Light-Water-Cooled Nuclear Power Reactors" acceptance criteria for long-term cooling. These requirements are summarized in Prairie Island USAR Section 6.2.1.1.

The NRC staff concludes that the proposed RICT for TS 3.5.2 Required Action A.1 does not impede accomplishing the ECCS specific safety functions because 100 percent of the ECCS flow is provided by the remaining OPERABLE ECCS train. The ECCS is designed to provide protection against postulated design-basis accidents caused by ruptures in the primary system piping. The coolant delivery rates are such that the ECCS performance under all LOCA conditions assumed in the Prairie Island design satisfies the requirements of 10 CFR 50.46 as discussed above.

3.1.2.5 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 has reviewed the licensee's proposed TS changes and supporting documentation. 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 defense-in-depth provided that the licensee identifies and implements compensatory measures as appropriate during the extended CT.

As discussed above in this SE, the NRC staff has further evaluated key safety functions in the proposed CT extensions and concluded that the changes are consistent with the defense-in-depth 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, 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 defense-in-depth 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 LAR is not proposing to change any quality standard, material, or operating specification. The LAR proposed to add a new program, “Risk Informed Completion Time Program,” in Section 5.5.18, “Administrative Controls,” of the TSs, which would require adherence 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, if sufficient trains remain operable to fulfill the TS safety function, the operability of the remaining train(s) ensures that the current safety margins are maintained. The NRC staff finds that if the specified TS safety function remains operable, sufficient safety margins would be maintained during the extended CT of the RICT Program. The NRC staff has evaluated specific proposed changes to the TS as described in Section 3.2 of this SE.

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 the risk of the configuration. During this increased CT, 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

As discussed above, the NRC staff finds that the design-basis analyses for Prairie Island 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 be 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 margin reductions are minimized by the implementation of the RICT Program. Based on the above, 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 5.5.18, "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 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, "Risk-informed Decision Making: Technical Specifications," of NUREG-0800, "Standard Review Plant for Review of Safety Analysis Reports for Nuclear Power Plants," dated March 2007 (SRP) (ADAMS Accession No. ML070380228), RG 1.177, and RG 1.174, Revision 1. 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, 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 NRC 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 LAR.

<u>PRA Acceptability</u>

The objective of the PRA acceptability review is to determine whether the Prairie Island PRA that was used to implement the RICT Program is of sufficient scope, level of detail, and technical adequacy for this application. 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-A, states that the PRA models should conform to the guidance in RG 1.200, Revision 1 (ADAMS Accession No. ML070240001). RG 1.200, Revision 2, clarifies that the current applicable ASME/ANS 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 supplement dated September 1, 2020, which included the response to the NRC staff's RAI, including industry peer review results and the licensee's self-assessment of the PRA models for internal events, including internal flooding, and fire, against the guidance in RG 1.200, Revision 2. The LAR stated that all external hazard events screen out, except for

seismic, as insignificant contributors to RICT calculations. The Prairie Island PRA model with modifications is used as the CRMP model.

The NRC staff's review of the key assumptions provided in the LAR is based on guidance and process described in NUREG-1855, Revision 1, "Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," dated March 2017 (ADAMS Accession No. ML17062A466). The NRC staff also requested clarification for two modeling assumptions that may not be included in the generic lists of PRA modeling assumptions, but which have been determined to be important for RICT Program implementation.

In RAI 9, the NRC staff noted the challenges of incorporating FLEX into a PRA model in support of risk-informed decision-making in accordance with the guidance in RG 1.200, Revision 2 and cited NRC guidance on implementing FLEX dated May 30, 2017 (ADAMS Accession No. ML17031A269). The licensee's response to RAI 9 stated that FLEX equipment or mitigating actions is not credited in the Prairie Island internal events, internal flooding, or fire PRA models that will be used for RICT calculations. The RAI response acknowledged that use of FLEX equipment may be credited in the future in accordance with NRC-accepted guidance.

In RAI 11, the NRC staff noted that, during its review of the Prairie Island National Fire Protection Association (NFPA) 805 LAR, it had requested additional information about the minimum joint human error probabilities (HEPs) used in the fire PRA. The licensee's response to an RAI for the NFPA 805 LAR, stated that it had updated its fire PRA to use a minimum joint HEP value of 1E-05 as noted in the NRC staff's approval of the NFPA 805 amendment for Prairie Island dated August 8, 2017 (ADAMS Accession No. ML17163A027). RAI 11 for this application requested clarification of whether the same fire PRA model was being used for the TSTF-505 application as was being used for the NFPA 805 application. The RAI requested that if the same model was not being used, to explain and justify the minimum joint HEP that will be used for the TSTF-505 application. The RAI also requested clarification of the minimum joint HEP treatment that is being used in the internal events PRA. The licensee's response to RAI 11 explained that during a recent fire PRA update the minimum joint HEP value was lowered to 1E-06. The response explained a review was performed to ensure that HEP combination assigned a minimum level of 1E-06 could be justified based on the level of dependence between HEPs or because of the HEP combination had a minimal contribution to risk. The response clarified that a minimum joint HEP of 1E-06 was used in the internal events PRA. The NRC staff finds this treatment to be acceptable because it meets NRC's interpretation of the fire human reliability analysis (HRA) guidance in NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," July 2012 (ADAMS Accession No. ML12216A104), and the general HRA guidance in NUREG-1792, "Good Practices for Implementing Human Reliability Analysis," dated April 2005 (ADAMS Accession No. ML051160213), on assignment of minimum joint HEP values.

Internal Events PRA (Including Internal Flooding)

The NRC staff review of the Prairie Island internal events (including internal flooding) PRA was based on the results of a full-scope peer review of the internal events PRA, two focused-scope peer reviews, and two facts and observations (F&Os) closure reviews described in LAR Enclosure 2. All peer reviews were performed using the guidance in the ASME/ANS RA-Sa-2009 PRA Standard and RG 1.200, Revision 2 to Supporting Requirement (SR) Capability Category (CC) II. The full-scope peer review of the internal events PRA was performed in November 2010 using the process described in NEI 05-04, "Process for

Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," dated November 2008 (ADAMS Accession No. ML083430462).

The first focused-scope peer review was performed in September 2012 on internal flooding events. The second focused-scope peer review performed in April 2014 on the PRA modeling of the Reactor Coolant Pump (RCP) seals. The two F&O closure reviews were performed in October 2017 and May 2019, by an Independent Assessment (IA) team, consistent with guidance in Appendix X to NEI 05-04/07-12/12-06 (ADAMS Accession No. ML16158A035), with clarifications and conditions provided in the NRC's acceptance letter dated May 3, 2017 (ADAMS Accession No. ML17079A427). One F&O, F&O SY-A17-01, remained following the second F&O closure review. NRC staff reviewed the disposition associated with F&O SY-A17-01 provided in LAR Enclosure 2, Table E2-1 concerning as-yet unreviewed and unaccepted credit taken in the PRA models for the RCP abeyance seal. In the disposition of the F&Os, as discussed in the LAR, the licensee stated that sensitivity studies demonstrated the RCP abeyance seal has minimal impact on the RICTs calculated for the SSCs in the RICT Program. However, the licensee's response to RAI 2.e, which requested further information on the sensitivity studies, stated that credit for the RCP abeyance seal PRA model will be removed from both the internal events and the fire PRA models.

The NRC staff reviewed the internal events F&O closure reports by the independent assessment team during the NRC staff's previous audit to support the review of the amendment requesting to adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" and to relocate specific TS surveillance frequency requirements to a licensee-controlled program. NRC staff found that the independent assessment team reviewed the open finding-level F&O to the PRA Standard SR applicable to the F&O at CC II. The NRC staff team found that a self-assessment of whether the resolution of a finding could constitute an upgrade of the PRA, as defined by the ASME/ANS RA-Sa-2009 PRA standard, and found that these determinations and their bases were reviewed by the independent assessment team. The NRC staff approved the request to adopt 10 CFR 50.69 for Prairie Island by amendment dated November 12, 2019 (ADAMS Accession No. ML19276F684).

In addition, unrelated to F&Os, LAR Attachment 5 stated the High-High Containment Pressure signal input to the MSIV closure logic will be added to the PRA prior to implementation of the RICT Program. This will facilitate performing a RICT calculation for TS LCO 3.3.2.D associated with ESFAS instrumentation.

Based on its review, the NRC staff finds that the internal events PRA including internal flooding, has been adequately peer reviewed against the current versions of the PRA standards (as referenced in Section 2.3.3 of this document) and RG 1.200, Revision 2, and that the remaining F&O is sufficiently resolved for the application. The NRC staff finds that the PRA, as modified as stated in the LAR and RAI responses, is technically acceptable and will model the as-built and as-operated plant and is, therefore, acceptable consistent with the guidance in RG 1.200, Revision 2.

Fire Events PRA

The NRC staff review of the Prairie Island fire events PRA was based on the results of a full-scope peer review of the fire PRA, a focused-scope peer review, and two F&O closure reviews described in LAR Enclosure 2. All peer reviews were performed using the guidance in the ASME/ANS RA-Sa-2009 PRA Standard and RG 1.200, Revision 2 to SR CC II. The

full-scope peer review of the fire PRA was performed in June 2012 using the process described in NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines." The focused-scope peer review was performed in March 2014 on the upgraded method to determine hot gas layer temperature. The first F&O closure review was performed in October 2017 on SRs with finding level F&Os from the 2012 full-scope review and the second F&O closure review was performed in May 2019 on findings against SRs that were not met or only met at CC I during the first F&O closure review. The two F&O closure reviews were performed consistent with guidance in the proposed Appendix X to NEI 05-04/07-12/12-06 with clarifications and conditions provided in the NRC's acceptance letter dated May 3, 2017. No fire PRA F&Os remained open following the second F&O closure review.

In RAI 2.a and b, NRC staff noted that LAR Attachment 5 provides two Implementation Items in Table A5-1 and states that, "[a]ll issues identified below will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the RICT Program." The first implementation item states that the RICT Program will use the same fire PRA model that was used to support NFPA 805 implementation. However, the NFPA 805 fire PRA model did not reflect the as-built, as-operated plant, but rather credited plant modifications and implementation items that NSPM committed to complete prior to implementation of the NFPA 805 program. Therefore, NRC staff requested confirmation ensuring that RICT calculations will use a fire PRA that reflects the as-built, as-operated plant. The licensee's response to RAI 2.a and b stated that all NFPA 805 modifications required by the NFPA 805 License Condition will be installed by the end of the third quarter of 2020 prior to the TSTF-505 LAR approval.

In RAI 2.c and d, the NRC staff noted that the second implementation item states that the High-High Containment Pressure signal input to the MSIV closure logic will be modeled in the internal events PRA prior to implementation of the RICT Program. The RAI requested confirmation that the implementation item applies to the fire PRA or explain why it is not and justify that the fire PRA model will be sufficient to support the RICT program. The licensee's response to RAI 2.c and d explained that there are no accident sequences in the fire PRA that would result in a High-High Containment Pressure signal that would require MSIV closure and therefore, no adjustment of the implementation item is needed. The response further explained there are fire accidents that can cause spurious actuations leading to LOCAs, but none of these accident scenarios require MSIV closure to mitigate the event. As discussed above, under internal events PRA, the response to RAI 2.e, stated that the unreviewed abeyance RCP seal PRA model will be removed from both the internal events and the fire PRA models.

During the audit, NRC staff reviewed the fire F&O closure reports and found that the independent assessment team reviewed the open finding-level F&O to PRA Standard SR 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 ASME/ANS RA-Sa-2009 PRA standard and found that these determinations were reviewed by the independent assessment team.

Based on its review, the NRC staff finds that the fire PRA has been adequately peer reviewed against the current versions of the PRA standard and RG 1.200, Revision 2 and that the F&Os have been adequately closed. The NRC staff finds that the fire PRA, as modified as stated in the LAR and RAI responses, is technically acceptable and will model the as-built and

as-operated plant and is, therefore, acceptable consistent with the guidance in RG 1.200, Revision 2.

PRA Technical Adequacy Conclusions

In Table A5-1, "RICT Program PRA Implementation Items" of LAR Attachment 5, as supplemented, the licensee identifies two items that will be completed prior to the implementation of the RICT Program:

- NSPM shall ensure that the fire PRA model used for the RICT Program reflects the as-built, as-operated plant using the same fire PRA model used to support National Fire Protection Association (NFPA) 805 implementation for both [Prairie Island] units prior to implementation of the RICT Program.

- NSPM shall ensure that the High-High Containment Pressure signal input to the MSIV closure logic is modeled in the [Prairie Island] PRA prior to implementation of the RICT Program.

The first item related to the NFPA 805, includes completing physical modifications to the plant, updating the fire PRA to model the physical modifications, and making some changes to the fire PRA models and methods. The second item includes making changes to the internal PRA models and methods (as indicated, the fire PRA is not affected by these model changes, as discussed in the above Fire Events PRA section of this SE).

Based on the NRC staff's review of the submittal, as supplemented, the NRC staff concludes that the Prairie Island PRA models for internal events, including internal flooding, and for fire events that will be 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 adequately conform to the applicable industry PRA standards for internal events, including internal flooding, and for fire events at an appropriate capability category, considering the 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, the Prairie Island PRA models were determined to be of sufficient technical adequacy to support implementation of the RICT Program. Therefore, the NRC staff finds that the request 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 Prairie Island PRA, 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 dated May 17, 2007 (ADAMS Accession No. ML071200238), notes that an LAR to implement a RICT program provide a discussion of the programs and procedures to ensure that the PRA models that provide the foundation for the CRMP model are maintained consistent with the as-built, as-operated plant. In the LAR, the term Real Time Risk model and CRMP model are both used and refer to the same one-line PRA model used in the CRMP. This SE uses CRMP for the program and CRMP model for the one-line PRA model throughout. Enclosure 7 of the LAR described a periodic update and review process for the PRAs that are used in the CRMP model. The NRC staff reviewed the

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 described that the update process is consistent with NEI 06-09-A. Section 2.2 of LAR Enclosure 7, "PRA Model Update Process" identifies that plant changes for incorporation into the PRA Model include, in part: (1) review and tracking of plant changes and discovered conditions for potential impact on the PRA models and the CRMP 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 nominally performed once every two fuel cycles; 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 implemented immediately.

The NRC staff concludes the licensee's PRA model update process is acceptable because it is consistent with NEI 06-09-A Sections 2.3.4 and 4.2, and RG 1.200, Revision 2.

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 constraint e. of TS Administrative Section 5.5.18:

> 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 approved for use with this program, 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 LAR has demonstrated that (1) the PRA has been reviewed using endorsed guidance and adequately resolved all identified issues, (2) a periodic update and review process has been established 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) RICTs will be calculated using NRC-accepted PRA methods. Therefore, the NRC staff concludes that a technically adequate PRA is implemented and will be maintained to support implementation of the RICT Program.

Scope of the PRA

Topical Report 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 in this Section, the PRA used for the RICT Program at Prairie Island includes contributions from internal events, including internal flooding, and fire events. In addition, the licensee provided a bounding estimate of the seismic CDF and LERF and included those CDF and LERF 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.

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.

The LAR provided an assessment of external hazard risk for the RICT Program in Enclosure 4, "Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models." Enclosure 4 states that this assessment is based on an update of the Prairie Island Individual Plant Examination of External Events (IPEEE) external hazard screening evaluation. The hazards assessed in LAR Enclosure 4, Table E4-2 are the same as those identified for consideration in non-mandatory Appendix 6-A of the ASME/ANS 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. According to the LAR, the following external hazards were evaluated:

- Aircraft Impact
- Avalanche
- Biological Event
- Coastal Erosion
- Drought
- External Flooding and Intense Precipitation
- Extreme Wind or Tornadoes
- 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
- Snow
- Soil Shrink-Swell
- Storm Surge
- Toxic Gas
- Transportation Accidents
- Tsunami
- Turbine-Generated Missiles
- Volcanic Activity
- Waves

LAR Enclosure 4, Section 2 states that consistent with NUREG-1855, Revision 1, 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, or(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, e.g., winds greater than design, seismic events greater than design-basis earthquake (DBE), etc. These beyond design basis conditions challenge the capability of the systems, structures, and components (SSCs) to maintain functionality and support safe shutdown of the plant.

- The second aspect addressed are the challenges caused by external conditions that are within the design basis, but still require some plant response to assure safe shutdown, e.g., high winds or seismic events causing loss of offsite power, etc. While the plant design basis assures that the safety-related equipment necessary to respond to these challenges are protected, the occurrence of these conditions nevertheless cause a demand on these systems that in and of itself presents a risk.

LAR Table E4-2 provided a screening disposition for each non-seismic external hazard 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 preliminary screening criteria and progressive screening criteria used and presented in LAR Table E4-3 is the same criteria presented SRs 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 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.

The LAR addressed the risk from seismic events and other external hazards in the context of this application in Enclosure 4. This enclosure provided 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 basis for exclusion of certain hazards from consideration in the determination of RICTs due to their insignificance to the calculation of configuration risk was also provided in the same enclosure, as discussed below. The LAR stated that the IPEEE external screening evaluation was updated to reflect current site conditions. External hazards considered were listed in Table E4-2 of Enclosure 4 to the LAR. The NRC staff finds that the list of external hazards considered by the licensee is consistent with the hazards listed in Appendix 6-A of the ASME/ANS RA-Sa-2009 PRA Standard, which is endorsed in RG 1.200, Revision 2.

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

LAR Enclosure 4, Section 3.0 states that RICT calculations will include a risk contribution from seismic events using a "seismic penalty" approach. The seismic penalty, commonly used in RICT calculations, and for estimation of seismic CDF (SCDF) is performed by a mathematical convolution of the seismic hazard and plant level seismic capacity curves. The seismic LERF (SLERF) is calculated using the convolution of the SCDF and containment seismic capacity. The licensee's proposed approach for including the seismic risk contribution in the RICT calculation is to add a constant SCDF and SLERF to each RICT calculation.

Section 3.3.5 of NEI 06-09-A states, in part, that for stations without external events PRAs, the station should apply one of three acceptable methods to determine external event risk. The second, often used, method is a reasonable bounding analysis which must be case-specific and technically verifiable and must be shown to be conservative from the perspective of RICT determination, which is the approach employed for the Prairie Island PRA. The licensee's proposed approach conservatively determined a SCDF estimate of 4.88E-07 per year (which is the updated value from the licensee's response to RAI 12). The SCDF estimate is based on the plant-specific seismic hazard curves developed in the licensee's March 27, 2014, response to the Near-Term Task Force (NTTF) Recommendation 2.1 (ADAMS Accession No. ML14086A628) and a plant-level high confidence of low probability of failure (HCLPF) capacity of 0.28g referenced to the controlling spectral frequency. The 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. The HCLPF parameters used for the Prairie Island SCDF estimate are the same as those shown for Prairie Island in Table C-2 of 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 NRC staff's RAI 12 noted that the estimated SCDF presented in the LAR Enclosure 4 was not based on the most current plant-specific seismic hazard curves. Therefore, the RAI requested an update of the SCDF estimate using current plant-specific seismic hazard curves developed in response to NTTF Recommendation 2.1. The licensee's response to RAI 12 provided an updated estimate of the seismic CDF (4.88E-07 per year). The response also explained that the updated SCDF value was based on a convolution of the plant-level HCLPF (as discussed above) and the current seismic hazard curve using PRA software (i.e., CAFTA 6.0b and FRANX 4.2). The response explained that the 10 Hertz (Hz) spectral frequency hazard curve was used because it produces the highest SCDF estimate.

LAR Enclosure 4, Section 3.3 describes the approach used to determine a SLERF "penalty" of 5 percent using the conditional large early release probability (CLERP) for internally initiated events with an adjustment for certain containment bypass events that would not be expected from a seismic event. RAI 12 noted that the LERF-to-CDF ratio for seismic events can be significantly higher than the same ratio for internal events due to the unique nature of seismically induced failures. Therefore, the RAI requested justification that the calculated LERF penalty is bounding. The licensee's response to RAI 12 stated that the seismic LERF was calculated by: "Convolving the same [Prairie Island] plant-level HCLPF seismic capacity (0.28g), composite variability (βc of 0.4) and the plant limiting HCLPF for containment integrity (0.30g), with the new site-specific hazard estimates for plants… ." The calculated SLERF value of 2.37E-07 per year, based on the 10 Hz seismic hazard curve, indicates that the approach for SLERF was based on the convolution of plant limiting HCLPF for containment integrity with the previously convolved SCDF, which is similar to the approach in the Joseph M. Farley, Units 1 and 2 (Farley), TSTF-505 safety evaluation dated August 23, 2019 (ADAMS Accession No. ML19175A243).) In the Farley TSTF-505 safety evaluation, NRC staff evaluated the inputs of plant-level HCLPF, composite beta factor, and the re-evaluated seismic hazard, dated October 16, 2015 (ADAMS Accession No. ML15287A092), and determined that the approach was acceptable because it was consistent with the corresponding values used for Farley in the Generic Issue (GI)-199 evaluation, which represented the most recent information on those parameters for Farley. Prairie Island's plant limiting HCLPF for containment integrity of 0.3g is not significantly higher than the plant-level HCLPF of 0.28g. The HCLPF of 0.28g is consistent with the corresponding value used for Prairie Island in the GI-199 evaluation dated August 2010 (ADAMS Accession No. ML100270639). In addition, the NRC staff's previous assessment of the Prairie Island re-evaluated seismic hazard dated December 15, 2015 (ADAMS Accession No. ML15341A162), concluded that the methodology used was acceptable and that the re-evaluation hazard appropriately characterized the site. As the Prairie Island's SLERF calculation uses an approach similar to that from Farley, which, as discussed above, the NRC staff previously found to be acceptable, the NRC staff finds that Prairie Island's approach to calculating the SLERF, and its results of 2.37E-07 per year, to be acceptable.

The NRC staff finds that during RICTs for SSCs credited in the design basis to mitigate seismic events, the proposed methodology captures the risk associated with seismically induced failures of redundant SSCs because the failure of 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 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.

The LAR states that the seismically induced loss of offsite power frequency is 5.48E-05 per year for Prairie Island, which is less than 6 percent of the total unrecovered loss of offsite power frequency in the internal events PRA for Prairie Island. The NRC staff evaluated the analysis and finds that the analysis adequately addresses the impact of seismically induced loss of offsite power and has an insignificant impact on the RICT Program calculations.

In summary, the NRC staff's review finds the licensee's proposal to use the seismic CDF contributions of 4.88E-07 per year and a seismic LERF contribution of 2.37E-07 per year, as an addition to the configuration-specific delta CDF and delta LERF from the internal events acceptable for Prairie Island's RICT Program because (1) the most current site-specific seismic hazard information for the Prairie Island is used; (2) an acceptable plant-level HCLPF value of 0.28g is used, consistent with the information in the Prairie Island GI-199 evaluation; (3) the SLERF was based on plant limiting HCLPF for containment integrity of 0.3g, which is conservative; and (4) adding 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 Winds and Tornado Hazards

LAR Enclosure 4, Table E4-2, presents the screening criteria used to disposition the risk for the extreme wind and tornado hazards. Table E4-2 indicates that criterion "PS2" (design basis for the event meets the criteria in the NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," dated September 1975 (ADAMS Accession No. ML081510817)) and criterion "PS4" (bounding mean CDF is <1E-06 per year) were 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-07 per year exceedance frequency is less than the Prairie Island design wind speed value.

The NRC staff's evaluation of the considerations of extreme wind and tornado hazard for Prairie Island finds that the extreme wind and tornado hazard has an insignificant contribution to configuration risk and can be excluded from the calculation of the proposed RICTs for Prairie Island because they either do not challenge the plant or they are bounded by the external hazards analyzed for the plant.

External Flooding Hazard

LAR Enclosure 4, Table E4-2, presents the screening criteria used to disposition the risk for the external flooding hazards. Table E4-2 indicates that criterion "PS1" (design basis hazard cannot cause a core damage accident) was used to screen the external flooding hazard and states that based on the flood hazard reevaluation report (FHRR), dated May 9, 2016 (ADAMS Accession

No. ML16133A041), and follow-up focused evaluation for the Prairie Island that external flooding does not challenge the current licensing basis or plant safety systems.

In RAI 13, NRC staff noted the statement in LAR Table E4-2 that during local intense precipitation, the site has "effective flood protection through the determination of Available Physical Margin and 'the reliability of protection features.'" The NRC staff also noted that the June 2014, NRC staff assessment report on the licensee's flooding walkdown report (ADAMS Accession No. ML14148A477) states that a deficiency in the flood response to ensure the power supply for portable sump pumps in case of loss of off-site power was identified. Accordingly, the NRC staff requested identification of the protective features credited for screening the external flooding hazard and justification that the reliability of these features are not affected by changes in plant configuration. The licensee's response to RAI 13 stated that the plant features credited for protection from external flooding consist of permanent plant features, such as building walls, doors, and flood bulkheads, as well as the inherent elevation of the plant site and temporary measures such as stop logs, berms, and sand bags. The response also indicated that none of these features are proposed for inclusion in the RICT Program and are not impacted by changes in plant configurations. The response also clarified that portable pumps could be deployed as additional protective measures, but only minimal water intrusion and local accumulation occur as a result of the probable maximum flood height.

The NRC staff's evaluation finds that the external flooding hazard has an insignificant contribution to configuration risk and can be excluded from the calculation of the proposed RICTs for Prairie Island because external floods either do not challenge the plant or they are bounded by the external hazards analyzed for the plant.

Other External Hazards

LAR Enclosure 4, Table E4 2, provided rationale for the insignificant impact of non-seismic external hazards and other hazards for Prairie Island. Notably, in RAI 14, the NRC staff noted the explanation in LAR Enclosure 4 for screening the snow hazard stated that the snow load that would result from the maximum recorded snowfall event in Minnesota would result in an estimated weight of 46.5 pounds per square foot, which is within the design basis 50 pounds per square foot. The NRC staff noted the small margin between the design basis roof live load and the load caused by the maximum recorded snowfall event. Accordingly, in RAI 14, the NRC staff requested justification for screening the snow hazard by showing the occurrence frequency of a snowfall event that would result in exceeding the design basis roof live load is acceptably low.

The licensee's response to RAI 14 explained that although areas of Minnesota have greater snowfall, the maximum recorded snowfall accumulation for the area near the plant is 19.9 inches. The response explained that based on a snow moisture density of 33 percent, the weight of 1 inch of snow accumulation is about 1.75 pounds per square inch. Assuming these values, 19.9 inches of snow would result in a roof load would of about 35 pounds per square foot which is well below the design limit for Prairie Island safety-related structures of 50 pounds per square foot.

The NRC staff's review of the information in the LAR and the supplemental information 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 for Prairie Island 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 licensee's approach for considering the impact of seismic events, non-seismic external hazards and other hazards for Prairie Island in the RICT calculations is acceptable because the approach includes a technically acceptable quantitative assessment of the seismic risk for Prairie Island consistent with the guidance in NEI 06-09-A and demonstrates 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 LAR, as supplemented, has satisfied the intent of RG 1.177, Revision 1 (Section 2.3.2), and RG 1.174, Revision 3 (Sections 2.3 and 2.5), and that the scope of the PRA model and the use of a bounding 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 modeled in the PRA. TSTF-505, Revision 2 also states 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 TS LCO Required Actions for which the RICT Program is proposed to apply, the following: (1) the SSCs included within the scope of the PRA models, or surrogate SSC or operator errors modeling that bounds 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 bases that justifies use of the PRA success criteria are justified and consistent with the RG 1.200, Revision 2 PRA review process; (3) the amendment implementation will update the PRA models to include the SSCs covered by the TS; and (4) the PRA models will be updated as part of the implementation to include the High-High Containment Pressure channels associated with TS LCO 3.3.2.D. The LAR further states that CCFs are appropriately addressed, the CRMP provides the capability to select the system as out of service in order to calculate a RICT, and the CRMP model is maintained consistent with the baseline PRA model.

System and Surrogate Modeling

LAR Enclosure 1, Table E1-1, as supplemented, in part, (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, explains how the PRA uses surrogate events that bound the function(s) of the TS LCO SSC(s).

In two instances, the LAR indicates that components modeled in the PRA will be used as surrogates for components not modeled in the PRA but are in the RICT Program. For TS LCO Condition 3.3.1.L, the LAR indicates that failure of the under-voltage channels will be used as a surrogate for failure of the under-frequency channels because they are logically equivalent in terms of impact. For TS LCO Condition 3.3.2.B, the LAR indicates that failure of the manual SI will be used as a surrogate for failure of the manual CI, because manual CI generates a CI signal with other signals and is therefore a conservative approach.

The NRC staff's RAI 1 noted that there appeared to be systems that could be shared and cross-tied between units that were credited in the RICT calculations. Therefore, the RAI requested explanation of how shared and cross-tied systems were modeled in the PRA and credited in the RICT calculations. The RAI also requested explanation of how initiators that can create a concurrent demand for shared or cross-tied systems at both units are addressed in the RICT calculations. The licensee's September 1, 2020, response to RAI 1 explained that the systems that are shared between units are the CL, External Circulating Water, Station and Instrument Air, and Control Room Ventilation systems. The response stated that these systems are designed to have sufficient capacity to supply both units concurrently which includes dual-unit initiating events.

The response further stated that the PRA models reflect the configuration changes to shared and cross-tied systems once initiating events have occurred, contingent on the initial configuration. The response explained that the systems that are cross-tied are the safeguards 4kV AC power and AFW systems. The response also explained that the CRMP model contains the top events for each unit, and therefore, shared or cross-tied systems that are taken out-of-service for one unit in the CRMP model will also impact the other unit. For the AFW, the PRA logic excludes the ability to cross-tie the motor-driven AFW pump for dual-unit initiating events and that for the safeguards 4kV AC power, the PRA logic assumes the power for the 121 CL Pump is unavailable for the Unit 1 D1 and D2 EDGs. The response also stated that the D1 and D2 EDGs have been shown to have sufficient capacity to supply the other 4kV AC safeguard loads for both units simultaneously. The response also stated that the D5 and D6 (Unit 2) EDGs have been shown to have sufficient capacity to supply the all 4kV AC safeguard loads for both units simultaneously. Accordingly, the NRC staff finds the RAI response indicates that the shared and cross-tied systems are appropriately modeled in the PRA and that appropriate unit specific RICTs can be calculated for each unit.

In RAI 8.a, the NRC staff requested an explanation of how analog I&C systems are modeled in sufficient detail to support the RICT Program.

The licensee's September 1, 2020, response to RAI 8.a stated that for each modeled I&C function, all channels are modeled in the PRA. Each instrument channel modeled in the PRA includes the transmitter and multiple bistables (depending on the functions and setpoints) and associated logic and actuation relays. The response explained that each bistable feeds train-specific relays that make up the logic (i.e., 1/2, 2/3, or 2/4) which in turn actuate one or more slave relays that actuate the actual equipment. The NRC staff finds that the PRA modeling of the analog I&C systems are performed in sufficient detail to support the RICT Program because all channels are modeled in the PRA and the risk important I&C risk contributors are modeled including the sensors/transmitters, the bistables, the logic relays, and the actuation relays.

In RAI 8.b, the NRC staff noted, regarding digital I&C systems, a lack of consensus industry guidance for modeling these systems in plant PRAs to be used to support risk-informed applications. In addition, known modeling challenges exist such as the lack of industry data for digital I&C components, the difference between digital and analog failure modes, and the complexities associated with modeling software failures including common cause software failures. Therefore, the RAI requested the results of a sensitivity study or identification of RMAs that will be applied to certain LCO Conditions during a RICT.

The response to RAI 8.b stated that the digital equipment installed at Prairie Island includes (1) the feedwater control system for power operation, (2) ATWS [anticipated transient without scram] Mitigating System Actuation Circuitry/Diverse Scram System (AMSAC/DSS), and (3) the safeguards bus load sequencers. The licensee explained that the feedwater safeguard functions (i.e., feedwater isolation, steam generator water level reactor trip) remain analog. The digital feedwater I&C is used for at-power operations, and therefore will not impact the change-in-risk calculations associated with RICT calculations. The response clarified that the AMSAC/DSS does not include any SSCs governed by the RICT Program and therefore will not impact the change-in-risk associated with RICT calculations.

The response explained that a sensitivity study was performed where the safeguards bus load sequencer individual failure and CCF events were increased by a factor of three to account for the uncertainty related to digital equipment failure rates and the increased potential for CCFs. The response stated that its impact on the recalculated RICTs was small. In the sensitivity case, all RICTs changed less than 4 percent except for two cases which had a 10 percent change corresponding to about 0.6 days. The NRC staff notes that a factor of three increase in failure rate is a relatively small factor and a corresponding 10 percent change is not negligible, based upon the NRC staff's engineering judgment, but, the licensee's evaluation reflects the state-of-the-practice and state-of-knowledge associated with digital equipment failure rates. Most of these RICTs exceed the 30-day backstop and the highest sensitivity cases are not expected to reduce these RICTs to the backstop value or less. The NRC staff recognizes the expectation of low failure rates and the reasonably small impact on the mostly long RICTs associated with these LCOs reflects the current state-of-the practice and that requests to exceed the 30-day backstop, if encountered, would provide the NRC staff the opportunity to reevaluate the digital PRA assumptions, if appropriate.

The NRC staff notes that the PRA configuration control program should follow any changes to the state of the practice, consistent with RG 1.200, Revision 2. Given these factors, the NRC staff finds that including a sensitivity study for the digital equipment failure equipment rate every time a RICT is calculated would have limited or no safety benefit and therefore is unnecessary.

Success Criteria

TSTF-505, Revision 2 does not allow for TS loss of function conditions (i.e., those conditions that represent a loss of a specified safety function or inoperability of all required trains of a system required to be operable) in the RICT Program. Additionally, the guidance in Item 11 in Section 2.3 of TSTF-505, Revision 2, states that "[t]he traveler will not modify Required Actions for systems that do not affect core damage frequency (CDF) or large early release frequency (LERF) or for which a RICT cannot be quantitatively determined."

LAR Enclosure 1, Table E1-1 appeared to the NRC staff to include TS LCO Conditions that may represent TS loss of function because the conditions may allow configurations that do not meet

the design basis success criteria indicated in Table E1-1. LAR Table E1-1 summarizes how the PRA success criteria differ from the design basis success criteria.

In RAI 6.a, b, and c, the NRC staff stated that it was not clear based on the information in the LAR Table E1-1 whether the design basis function of the applicable equipment could be satisfied for all conditions for which a RICT is proposed. In RAI 6.a, the NRC staff noted that the comment column in TS LCO 3.6.3.A and LCO 3.6.3.C both include the statement that "[o]nly penetrations that can contribute to LERF are modeled." Therefore, the NRC staff requested explanation of whether a RICT will be applied to penetrations that do not contribute to LERF and, if so, how the RICT, consistent with the guidance in TSTF-505, Revision 2, is calculated. The licensee's response to RAI 6.a explained that a RICT will be applied to non-modeled containment penetrations that are screened because they are too small to contribute to LERF. The response clarified that the increase in risk for these RICTs will be based only on the seismic penalty factors, if no other equipment is out of service. The response further stated that a RICT will not be applied to any non-modeled penetrations that are large enough to result in a large early release when failed open but are screened due to being locked closed.

Application of a RICT to non-modeled small penetrations is acceptable because any increase in CDF associated with the unavailability of isolation of these small containment penetrations, is expected to be negligible and there is, by definition, no impact on LERF. Not applying a RICT to larger, but screened out penetrations, is consistent with NEI-06-09-A that unmodeled SSCs that might impact risk metrics but are not included in the calculations should be excluded from the RICT Program.

The NRC staff noted in RAI 6.b that the success criteria presented for TS LCO 3.7.1 in LAR Table E1-1 is "Five of five MSSVs [main steam safety valve] per SG" and the PRA success criteria is "One of five MMSVs per SG when associated [SG] PORV and steam dump not available." Therefore, the RAI requested clarification of why one inoperable MSSV is not considered a TS loss of function. The licensee's response to RAI 6.b explained that five-of-five MSSVs per SG are required for a trip from full power with no other means of steam relief for a short period of time when the decay heat load is highest. The response confirmed that a single out-of-service MSSV would represent a loss of TS safety function during the period of time when the decay heat load is the highest. Therefore, TS LCO 3.7.1 was removed from the proposed RICT Program as shown in the updated Table E1-1, Table E1-2, and TS markups in Attachments 1, 2, and 3 provided in the response.

RAI 6.c noted in LAR Table E1-1 for LCO 3.6.5.C and LCO 3.6.5.D the Containment Spray (CS) and Containment Cooling Fan Coil Units (FCUs) are not modeled in the PRAs based on the results of thermohydraulic analyses showing that the unavailability of the CS trains or FCUs have no impact on CDF or LERF. The RAI also noted the statement in LAR Section 2.6 explaining that, in part, "[a]dverse impacts caused by operation of the CS are considered." Given that these two statements seemed to be contradictory, the RAI requested: (1) clarification of what modeling exists in the PRAs of these two system and what impact the modeling has on CDF and LERF; (2) summarization of the minimum equipment needed to fulfil the design basis function associated with these LCO conditions to confirm that LCO 3.6.5.C and LCO 3.6.5.D do not represent loss of function conditions; (3) explanation of how the risk increase will be calculated when the a CS train or FCU is taken out of service; and (4) description of the cited hydraulic analysis and justification that it is applicable regardless of plant configuration.

In response to RAI 6.c, the licensee confirmed that the containment cooling function (i.e., CS and FCUs) are not modeled in the PRAs. The RAI response explained, however, that potential

adverse consequences of operation of the CS (e.g., reduced time to realign to recirculation) are included in the modeling for completeness. The response further clarified that the minimum equipment needed to fulfil the TS design basis function for LCO 3.6.5 is one-of-four FCUs and not two-of-four as reported in LAR Table E1-1. The response provided an updated version of LAR Table E1-1 making this correction. NRC staff finds that LCO 3.6.5.C an LCO 3.6.5.D do not represent a TS loss of function because just two of four FCUs are inoperable in each case and only one FCU is needed to fulfill the TS design basis function. The licensee's response explained that applicable Modular Accident Analysis Program (MAAP) runs were reviewed to confirm that PRA scenarios that did not meet the criteria for LERF would not meet the criteria even without operation of a CS train or FCU. The response stated that the LERF criteria would not be met for any configuration allowed by the RICT Program. Based on the description provided in the RAI response, the NRC staff finds the proposed treatment of the CS and FCU system in the RICT calculations acceptable because the unavailability of a CS train or FCU is appropriately reflected in the RICT calculations and concludes that LCO 3.6.5.C and LCO 3.6.5.D do not represent a TS loss of function because just one of four FCUs is needed to fulfill the TS design basis function.

In RAI 17.1, 17.2, 17.3, and 17.4, the NRC staff requested justification for a number of LCOs that may represent TS loss of function conditions, which is not allowed by TSTF-505, Revision 2. These conditions may have the potential to defeat the design basis success criteria presented in LAR Table E1-1. RAI 17.1 noted that TS LCO 3.3.1, Reactor Trip System Instrumentation, Condition M (One Reactor Coolant Pump Breaker Open channel inoperable) may result in TS loss of function when thermal power is greater than the P-7 interlock [setpoint] and when the thermal power is less than the P-8 interlock [setpoint]; and that Condition O (One Turbine Trip channel inoperable) may result in TS loss of function. The licensee's response to RAI 17.1 acknowledged that LCO 3.3.1.M results in a TS loss of function when thermal power is greater than the P-7 interlock and less than the P-8 interlock. The response provided a revised proposed TS to include a note for Condition M excluding use of a RICT when thermal power is greater than the P-7 interlock and less than the P-8 interlock. The response clarified that the TS design basis success criteria for LCO 3.3.1.O presented in the LAR is incorrect and that the correct TS design basis success criteria should be "Two of Two Turbine Stop Valve Closure channels or two of three Low Autostop Oil Pressure channels when [the thermal power is] above the P-9 interlock." The response provided the corrected TS design basis success criteria in an updated LAR Enclosure 1, Table E1-1. The NRC staff finds, given the cited TS and LAR corrections, that TS LCO 3.3.1.M and TS LCO 3.3.1.O do not represent TS loss of function because the conditions do not defeat the TS design function.

RAI 17.2 noted that TS LCO 3.3.2, Condition B (One channel or train inoperable), may result in TS loss of function for CS Manual Initiation; and Condition F (One channel or train inoperable) may result in TS loss of function for Steam Line Isolation Manual Initiation. The licensee's response to RAI 17.2 acknowledged that LCO 3.3.2.B does represent a TS loss of function for CS Manual Initiation. The response provided a revised proposed TS and updated LAR Enclosure 1, Table E1-1 that excludes the use of the RICT for the CS Manual Initiation function. The response clarified for LCO 3.3.2.F that the TS design basis success criteria for LCO 3.3.2.F presented in the LAR is incorrect and that the correct TS design basis success criteria should be "One of two Manual Initiation channels (switches and associated logic)." The response provided the corrected TS design basis success criteria in an updated LAR Enclosure 1, Table E1-1. The NRC staff finds, given the cited corrections, that TS LCO 3.3.2.B and TS LCO 3.3.2.F do not represent TS loss of function because the conditions do not defeat the TS design function.

RAI 17.3 noted that TS LCO 3.3.4, 4kV Safeguards Bus Voltage Instrumentation, Condition C (Required Action and associated completion time of Condition A or B not met, or Function a or b or both with three channels per bus inoperable, or one required automatic load sequencer inoperable) may result in TS loss of function when Function a or b or both with three channels per bus are inoperable. The licensee's response to RAI 17.3 explained that loss of TS function would not occur unless the three channels of Function a and b were inoperable on both buses for the same unit, but that would render the automatic load sequencer for both buses inoperable. Losing the automatic load sequencer for both buses defeats one of the criteria listed in TS LCO 3.3.4.C (i.e., just one load sequencer is inoperable), and therefore, the RICT could not be entered. The NRC staff finds that TS LCO 3.3.4.C does not result in TS loss of function because if the instrumentation functions for both buses are unavailable, then the automatic load sequencers for both buses are also inoperable and a RICT cannot be entered.

RAI 17.4 noted that TS LCO 3.7.8, Condition A (no safeguards CL pumps operable for one train), may result in TS loss of function given a statement in USAR Section 6 (ADAMS Accession No. ML20118D368) explaining that "one CL pump is required to operate during the recirculation phase to cool the recirculation flow and containment atmosphere in the unit suffering the accident and provide the necessary cooling for the other unit." The licensee's response to RAI 17.4 explained that Condition C represents the condition in which one train is inoperable, but the other train is operable with its applicable safeguards pump also operable. The NRC staff finds that TS LCO 3.3.4.C does not result in TS loss of function because one CL pump is operable meeting the TS design function.

Common-Cause Modeling

Section 3.3.6 of NEI 06-09-A states that for all RICT assessments of planned configurations, the treatment of CCFs in the quantitative configuration risk management tools may be performed by considering only the removal of the planned equipment and not adjusting CCF terms. The guidance in RG 1.177, Appendix A, Section A-1.3.1.1, states: "If the component is down because it is being brought down for maintenance, the CCF contributions involving the component should be modified to remove the component and to only include failures of the remaining components (also see Regulatory Position 2.3.1 of Regulatory Guide 1.177)."

In RAI 4, the NRC staff requested an explanation for how CCF contribution is included in the PRA models used for the RICT Program and how the PRA models are adjusted when a component from a CCF group (such as a group of three components from three trains that provide 100 percent of the required function) is removed for planned preventive maintenance. The licensee's response to RAI 4 explained that common cause basic events are explicitly modeled in the PRAs with the corresponding independent basic events using the multiple Greek letter (MGL) method. The response explained that no adjustments will be made in the CCF factors of the CRMP model and that this approach is conservative because CCF combinations that no longer apply are retained.

In general, the NRC staff notes that the CCF contribution from the out-of-service component is conservatively retained in the following ways: (1) the independent failure rate used in the PRA models includes both independent and dependent failure events (i.e., the dependent failures should be subtracted from the total population of failures to calculate the independent failure rate) and (2) the CCF event probabilities that include the out-of-service component are retained. The NRC staff also notes, however, that this simplification produces both conservative effects (as described) and non-conservative effects (based on the actual MGL factors applied). The CCF probability estimates contain a degree of uncertainty and retaining precision in the

calculation of these estimates using a more refined approach will not necessarily improve the accuracy of the results. Therefore, the NRC staff concludes that the method is acceptable because the calculations reasonably include CCFs after removing one train for maintenance consistent with the accuracy of the estimates.

Concerning entering TSs for emergent conditions, consistent with TSTF-505 Revision 2, the administrative TS requirement (TS 5.5.18, item d) specifies that in an emergent condition, if the extent of condition for the inoperable SSC is not complete prior to exceeding the ACTION allowed outage time, then the RICT Program will account for the increased possibility of CCF. The RICT Program will account for increased possibility of CCF by either (1) numerically accounting for the increased possibility of CCF in the RICT calculation or (2) implementing RMAs, not already credited in the RICT calculation, 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.

In RAI 5, the NRC staff requested a description of how the numerical adjustment of CCFs for the increased possibility of CCF for emergent failures will be performed if option 1 is selected. In its response to RAI 5, the licensee stated that if option 1 is used, that the increased CCF probability for the remaining in-service components in a common cause group will be calculated in accordance with RG 1.177, Revision 1. The response stated that in this case, all probabilities for the existing CCF events in the affected common cause group will be re-calculated by dividing those probabilities by the failure probability of the failed component which has the effect of raising the CCF probability to account for the failure. The NRC staff finds this approach to numerically adjusting the increased CCF probability for the remaining in-service components in the common cause group (i.e., use of option 1) acceptable because it is consistent with the guidance in RG 1.177, Revision 1, Appendix A, Section 1.3.2.1.

CRMP Model

The CRMP model is the PRA model used by the CRMP to perform the RICT calculations. The CRMP provides a user interface which supports the RICT Program by providing a method to evaluate the plant configuration.

LAR Enclosure 8, “Attributes of the Configuration Risk Management Model,” describes the necessary changes to the peer-reviewed baseline PRA models for use in the CRMP software to support RICT calculations that: preserves the CDF and LERF quantitative results; maintains the quality of the peer-reviewed PRA models; and correctly accommodates changes in risk due to configuration-specific considerations.

Enclosure 8 of the LAR explains that the peer-reviewed internal events and internal flooding models are integrated into one PRA model and the fire events PRA model are maintained as separate models. However, for RICT Program implementation, these baseline models are incorporated into the CRMP software and modified/adjusted as follows for use in configuration risk calculations: (a) the unit availability factor is set to 1.0 (unit available); (b) maintenance unavailability is set to zero/false unless unavailable due to the configuration; (c) mutually exclusive combinations, including normally disallowed maintenance combinations, are adjusted to allow accurate analysis of the configuration; and (d) for systems where some trains are in service and some in standby, the CRMP model addresses the actual configuration of the plant including defining in-service trains as needed. These adjustments are the same as those used for the evaluation of risk under the Maintenance Rule program (i.e., 10 CFR 50.65(a)(4)). The CRMP software is designed to quantify the unit-specific configuration for both internal events,

including internal flooding, and fire events, and includes the seismic risk contribution when calculating the RMAT and RICT. The LAR explained that CRMP software will be used to facilitate all configuration-specific risk calculations and the baseline PRA models will be modified to create a single top CRMP model.

LAR Enclosure 8 stated that maintenance of the CRMP model and transition from the PRA models to the CRMP model will be controlled and documented in accordance with fleet procedures and include a process for identification and disposition of model errors. The LAR stated that the plant procedures specify that an acceptance test is performed after every CRMP model update to ensure that the software works as intended and that cutset quantification results are reasonable. The plant procedures and this test will verify proper translation of the baseline PRA models and acceptance of changes made to the baseline PRA models into the CRMP model.

In RAI 15.a, the NRC staff requested a description of how any changes in environmental conditions due to seasonal variations are accounted for in the CRMP model for use in RICT calculations. RAI 15.a also requested a discussion of impacts on the plant response model and the initiating event frequencies. The licensee's response to RAI 15.a explained that seasonal based and cycle variations are accounted for in the CRMP model through applying the correct alignment or activities, such as the following:

- accounting for the correct number of CL pumps which varies from one to three pumps depending on river temperature and system demand,

- accounting for fine versus the coarse traveling screen modes which varies throughout the year based on environmental permits,

- adjustment of exposure (i.e., cycle) time which varies the success criteria for pressure relief during an ATWS, and

- accounting for outside grid impact such as when transmission lines are out of service or there is switchyard maintenance by increasing the LOOP frequency in the CRM model.

The LAR stated that heat-up analyses show that active cooling is needed in the Main Control Room but not elsewhere in the plant and are based on using the conservative temperature inputs across the year. Therefore, no adjustment in the PRA is needed for these treatments.

In RAI 15.b, the NRC staff requested confirmation that out-of-service equipment will be reflected as appropriate in the CRMP model initiating events as well as the plant response model. The licensee's response to RAI 15.b confirmed that out-of-service equipment will be properly reflected in the CRMP model initiating event models through the Support System Initiating Events fault trees. Accordingly, when a component is out-of-service, it affects the Support System Initiating Events fault trees as well as the mitigating system fault trees.

In RAI 15.c, the NRC staff requested a description of the process that will be used to maintain the accuracy of any pre-solved cutsets with changes in plant configuration. The licensee's response to RAI 15.c explained that no pre-solved cutsets will be used in the Prairie Island CRMP model used in the RICT calculations.

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.

PRA Modeling Conclusions

The NRC staff reviewed the information provided in the LAR, as supplemented, 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 intent of RG 1.177, Revision 1 (Section 2.3.3), and RG 1.174, Revision 3 (Section 2.3), have been satisfied, 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, if necessary, applicable RMAs to limit the impact of these uncertainties. Enclosure 9, "Key Assumptions and Sources of Uncertainty" of the LAR discussed key assumptions and sources of uncertainty.

The LAR stated that the Prairie Island PRA models were evaluated to identify the key assumptions and sources of uncertainty for this application consistent with the RG 1.200 definitions. The evaluations were completed 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.

LAR Enclosure 9, Section 2 stated that the internal events PRA uncertainty analysis was performed based on guidance in NUREG-1855, Revision 1. The LAR explained that plant specific assumptions and sources of modeling uncertainty identified in the internal events PRA notebooks were considered, as well generic sources of uncertainty from the Electric Power Research Institute (EPRI) Technical Report (TR)-1016737, "Treatment of Parameter and Modeling uncertainty for Probabilistic Risk Assessments." The LAR explained that the three sources of uncertainty defined in NUREG-1855 are considered in the RICT Program: parametric uncertainties, modeling uncertainties, and completeness uncertainties. Parametric uncertainties are specifically addressed in the PRA quantification to develop probability distributions for CDF and LERF and determine mean estimates for the risk values. In regard to the fire PRA, the licensee states, in LAR Enclosure 9, Section 3, that it used guidance from NUREG-1855, Revision 1 and current guidance for fire PRA including NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities" (ADAMS Accession Nos. ML052580075 and ML103090242), to address the fire PRA uncertainty analysis. The LAR explained that plant-specific assumptions and sources of modeling uncertainty identified from the fire PRA were considered, as well as generic sources of uncertainty from Appendix B of EPRI TR 1026511, "Practical Guidance of the Use of Probabilistic Risk Assessment in Risk-informed Applications with a Focus on the Treatment of Uncertainty," dated December 2012.

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. In RAI 16, the NRC staff stated that it was not clear what specific process and criteria were used to come to this conclusion. Therefore, the RAI requested description of the specific process and criteria (including those associated with plant specific features, modeling choices, and generic industry concerns) that were used to screen uncertainties from an initial comprehensive list of assumptions and sources of PRA modeling and to discuss sensitivity studies that were performed and their results showing that the key assumptions or sources of uncertainty have no impact on RICT calculations. The RAI also requested description of plant procedures and practices that were used to support identification and dispositioning of PRA modeling sources of uncertainty. The licensee’s response to RAI 16.a explained that the process for evaluation of uncertainties follows the guidance illustrated in Figure 4-1 of EPRI TR-1016737. The response explained that in compiling a comprehensive listing of plant-specific and generic sources of uncertainty that certain issues were not included if: (1) the uncertainty notebook already provided justification that the uncertainty did not need to be further evaluated (e.g., based on its negligible contribution to risk or the fact that best-estimate modeling was used) or (2) the generic source of uncertainty did not apply to Prairie Island (e.g., because of plant design or sufficient PRA modeling). The response explained that the resulting list of candidate key assumptions and sources of uncertainty were evaluated for its impact of the RICT Program using the following considerations:

- the candidate uncertainties were determined to be resolved in the current model of record either through modeling improvements or through incorporation of a documented basis in the PRA notebooks (e.g., by showing the uncertainty can be screened in accordance with the ASME/ANS PRA standard or by showing that the impact on the PRA results are negligible),

- sensitivity studies were performed that confirmed the impacts from the candidate uncertainties were negligibly small and considered not to impact the RICT Program,

- the candidate uncertainties were represented through conservative PRA modeling that would result in more limiting RICTs than would be calculated using best estimate modeling,

- the candidate uncertainties were represented through current industry-accepted approaches,

- the candidate uncertainties would not impact CDF or LERF (e.g., post-accident containment phenomenology uncertainty that would only impact late releases from containment), and

- the candidate uncertainties were determined to be not be uncertainties through use of deterministic analysis (e.g., thermal-hydraulic analyses or battery load calculations). The licensee provided several examples that included confirmation through analyses of factors such as mission times and success criteria.

The licensee’s response also described an element of the PRA maintenance and update program in which a PRA change form is created and tracked in an electronic database for all

issues that could impact the PRA models, including resolving concerns about PRA modeling assumptions and uncertainties.

In RAI 7, the NRC staff stated that during the audit, a review of the results of the uncertainty analysis identified a few sources of uncertainty that may have the potential to impact the RICT calculations. RAI 7.a noted the use of failure rates referred to in the calculational notes as having been developed using sources and means other than the industry data presented in NUREG/CR-6928 "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants" (ADAMS Accession No. ML070650650). Therefore, the RAI requested an explanation of what kinds of failures were included and whether the failure rates developed as part of the calculational notes were for non-typical components. The RAI also requested justification for treatment of the uncertainty associated this group of failures and associated failure rates.

The licensee's response to RAI 7.a explained that when failure rates from NUREG/CR-6928 could not be accurately used to reflect a failure, then other sources of failure data were used, or the estimates were developed based on plant-specific information. The response provided examples of such failures and also separately listed non-NUREG/CR-6928 failure data sources that included recognized sources such as datasets and guidance from Westinghouse, the Institute of Nuclear Power Operations, and Idaho National Laboratories. The NRC staff notes that PRA Standard SR DA-C1 stipulates generic parameters estimates come from recognizable sources. The response also explained that a sensitivity study was performed on the failure rates developed entirely on plant-specific information (e.g., plant operator logs, weather data and historical events) by increasing the failure probabilities by a factor of 3. The response referred to these events as "special events" and stated that the results of the sensitivity study showed a small impact on the calculated RICTs and that a majority of the RICTs for the sensitivity case did not change. However, five of the longer RICTs (i.e., RICTs with a 25-day duration) changed 4 percent, and three of the shorter RICTs changed by 8 to 11 percent. The NRC staff finds the use of failure rates other than those provided in NUREG/CR-6928 acceptable because either recognizable sources consistent with PRA Standard SR DA-C1 were used or the impact of a sensitivity study showed that the uncertainty of using the best estimate plant-specific information has a small impact on the calculated RICTs.

In RAI 7.b, the NRC staff noted that components without cable routing were assumed failed unless further analysis was performed to assure the systems are not compromised by transient fires. The RAI also noted that although this modeling treatment produces a conservative estimation of fire CDF and LERF, it can underestimate the change-in-risk determined in the RICT calculations by masking risk which results in the underestimation of associated completion times. Therefore, the RAI requested identification of the systems or components that are assumed to always be failed (or are not included) in the fire PRA due to lack of cable tracing and justification that this treatment has an inconsequential impact on the RICT calculations. The licensee's response to RAI 7.b explained that the systems or components that are assumed to always be failed in the fire PRA due to lack of cable tracing are the main feedwater system, Station and Instrument Air, four non-safety-related panels that only supply the AMSAC, four AC buses that support main feedwater and RCP operations, and the screenhouse well pump that provides non-safeguards bearing water supply to the 22 diesel-driven cooling water pump, the circulating water pumps and the 11 and 21 CL pumps. The response stated that these systems and components are not associated with LCOs proposed for the RICT Program and assuming their failure will not have a significant impact on the RICTs calculated for in-scope LCOs. The response also stated that review of the significant and non-significant accident sequence cutsets did not identify these systems and components as significant contributors to CDF. The

response stated that an additional sensitivity study was performed on the instrument air system in which it was assumed instrument air would always be available instead of always failed following a fire. This study resulted in a decrease in CDF and LERF of less than 1 percent. The NRC staff concludes that the impact of conservatively modeling systems and components without cable tracing to always be failed following a fire has been systematically evaluated by the licensee. The results of the licensee's evaluation indicate that the assumption that the systems are always failed after a fire is not significant to risk.

In RAI 7.c, the NRC staff noted that a source of modeling uncertainty concerns diversion flow paths that were not modeled for the residual heat removal (RHR) system because the failure of an RHR train is dominated by other train failures "by more than 2 orders of magnitude." The RAI observed that the cited diversion failures could potentially have a more significant contribution to the failure of systems that interface the RHR system. Therefore, the RAI requested explanation of the impacts that the cited diversions could have on systems that interface the RHR system and justification that exclusion of these failure mechanisms from the PRA models have an inconsequential impact on RICT calculations.

The licensee's response to RAI 7.c discussed the possible diversion possibilities cited in the uncertainty analysis for the three diversion pathways. The response confirmed through explanation supported by flow diagrams that the impact of these diversions on interfacing system would not contribute to CDF or LERF. In the first diversion pathway, failure of the RHR heat exchanger cross-tie valves that divert flow to the SI crossover or CS suction lines, the licensee identified multiple possibilities, which can result in this diversion. One possibility is that there is a diversion to the SI suction lines. In this event, the water will still be injected into the RCS through the SI lines via the cold leg. Another possibility is diversion to the CS suction lines either to the (1) CS spray header, (2) back to the refueling water storage tank, or (3) through the caustic addition lines. The response explained that if the diversion is to the CS header, then the impact on CS operation is negligible. The response also explained that if the diversion is back to the refueling water storage tank through a failed check valve or to the caustic addition Standpipe and Surge Tank, then the impact does not create another system failure that would impact CDF or LERF. The second diversion flow path is failure of RHR heat exchanger cross-tie valves that divert flow to the letdown line. For this case, the licensee stated that this diversion creates no intersystem impacts of concern (i.e., impact of the RHR on the chemical volume control system or impact of the chemical volume control system on the RHR system). In the third case, component cooling diversion through non-operating component cooling water system pumps, the licensee stated that the component cooling water system is a closed system, and therefore, there would be a negligible decrease in component cooling water flow to cooling loads supported by the component cooling water system and there would be no intersystem impacts. The NRC staff finds the treatment of this source of uncertainty acceptable because the impact of the cited potential diversions RHR is insignificant and because the licensee reviewed each diversion possibility cited in the uncertainty analysis and explained why the impact of those diversion on interfacing systems do not affect CDF or LERF.

In RAI 7.d, the NRC staff noted that credit for external vessel (ex-vessel) cooling to prevent core melt from escaping the vessels based on "realistic" MAAP modeling was identified as a source of uncertainty. Therefore, the RAI requested description of the MAAP modeling, assumptions, and results that justifies this credit. The licensee's response to RAI 7.d called the analysis a "best-estimate MAAP analysis" and stated that the analysis validated the conclusion that core debris would be cooled if RWST contents were discharged into containment. The response explained, however, that a 0.1 vessel failure probability is assumed even with ex-vessel cooling given concerns about the lower portion of the reactor vessel. The licensee's response

demonstrated that the modeling treatment had a small impact on LERF and would not impact the RICT calculations. The response showed that in approximately 99 percent of the total LERF scenarios that ex-vessel cooling was not credited and, in one scenario, (contributing to 1.2 percent of the total LERF) that ex-vessel cooling was credited but is assumed not to arrest core melt. Additionally, the NRC staff observes that the impact of this assumption on the baseline PRA LERF and the LERF associated with plant configurations allowed in the RICT Program is about the same, and so, the impact of the assumption on the change-in-risk calculations associated with RICT calculations is minimal. Therefore, the NRC staff finds the treatment of this uncertainty acceptable because it has a minimal impact on LERF and on RICT calculations.

LAR Enclosure 9, Table E9-1 provides an assessment of three key assumptions and sources of modeling uncertainty associated with translation of the PRA models to the CRMP model. The NRC staff reviewed the disposition to these three uncertainty impacts and found them to be acceptable because (1) in the case of optimizing the logic structure, the resulting logic will be equivalent to the original and acceptance testing will be performed for every CRMP model update, (2) in the case of seismic modeling the impact is conservative on the RICT calculations, and (3) in the case of plant availability the plant availability is set to 1.0 in the CRMP model, which has a conservative impact on the RICT calculations.

The impact of addressing the state of knowledge (SOKC) in the propagation of parametric uncertainty is an important source of uncertainty and is addressed in the next section of this SE on PRA results and insights.

The NRC staff's review indicates that an adequate assessment was performed to identify the potential sources of uncertainty, and the identification of the key assumptions and sources of uncertainty was appropriate and consistent with the guidance in NUREG-1855, Revision 1, and associated EPRI TR-1016737 and EPRI TR-1026511. Therefore, the NRC staff finds that 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) has been adequately addressed, and that the identification of assumptions and treatment of model uncertainties for risk evaluation of extended CTs is appropriate 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. Topical Report 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 4A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated April 2011 (ADAMS Accession No. ML11116A198). 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 increase below 1E-5/year for changes to CDF; (2) a total risk increase 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. Enclosure 5 of the LAR indicated 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.

In RAI 3, the NRC staff noted that based on RG 1.174 and Section 6.4 of NUREG-1855, Revision 1, for a CC II risk evaluation, the mean values of the risk metrics (total and incremental values) need to be compared against the risk acceptance guidelines. However, under certain circumstances, a formal propagation of SOKC uncertainty may not be required if it can be demonstrated that the risk results are well below the acceptance guidelines. Accordingly, RAI 3 requested that the mean values of the risk metrics (i.e., including the impact of SOKCs) be provided to compare to the point estimates (i.e., excluding the impact of SOKCs) provided in the LAR and that will be used in the RICT Program. The licensee’s response to RAI 3 provided the mean value of the total CDF and LERF for the internal events and fire PRAs and the updated seismic risk contribution from the response to RAI 12. The licensee’s response indicated that the impact of the SOKC was calculated for random equipment failure categories, HRA, special basic events, initiator frequency (including fire initiators), and circuit failure likelihood analysis.

As shown in Table 1 below, these total mean values based on the current model of record demonstrate that the estimated total mean 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. Comparing these mean values with the point estimate values reported in the LAR (not provided below) indicates that the SOKC increases the risk estimates by less than 10 percent. All mean values are much less that 10 percent below the guideline values indicating that use of the point estimates in the RICT Program will provide confidence that all the RG 1.174 guidelines will be adequately met.

Table 1 Total Baseline Risk for Prairie Island

	Internal[1] Events Risk	Fire Risk[1]	Seismic Risk[2]	Total Risk
Unit 1 CDF	1.36E-05	6.65E-05	4.88E-07	8.06E-05
Unit 1 LERF	2.26E-07	9.66E-07	2.37E-07	1.43E-06
Unit 2 CDF	1.27E-05	6.63E-05	4.88E-07	7.95E-05
Unit 2 LERF	1.90E-07	9.28E-07	2.37E-07	1.36E-06

Notes:

1. Based on updated internal events (including internal flooding) and fire PRA CDF and LERF values provided in response to RAI 3.
2. Conservatively calculated seismic “penalty” values based on the values provided in response to RAI 12.

NEI 06-09-A in the RICT Program of TS 5.5.18, has been incorporated and therefore the RICT can be calculated consistently with the criteria and assessed in 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 RICT Program is consistent with NEI 06-09-A guidance and is acceptable.

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, “Program Implementation,” 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 personnel using the RICT should provide confidence that the RICT Program will be appropriately applied.

The means for demonstrating the technical acceptability of the PRA models include assessment against the ASME/ANS 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, Revision 2,” and Enclosure 7, “PRA Model Update Process,” of the submittal. According to Enclosure 8, “Attributes of the Configuration Risk Management Model,” future changes made to the baseline PRA model, changes made to the baseline PRA model for translation to the online CRMP model, and 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, 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 LAR 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 pre-planned 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 implement appropriate programmatic and procedural controls for the 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 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 staff implementing the RICT Program. The LAR identified 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, and 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.

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 acceptable administrative controls have been proposed 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.

Topical Report 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. The guidance in NEI 06-09-A specifies that if the instantaneous CDF and LERF limits are exceeded for emergent conditions, then implementation of RMAs is required. 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 (the RMAT). 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 increase 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 power source, a battery charger, and an RHR pump. 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. The development of RMAs is performed in a graded manner and considers RMAs developed for the Maintenance Rule, 10 CFR 50.65(a)(4) program. Besides the three kinds of RMAs cited above, the LAR stated it uses general, configuration-specific, and common cause RMAs. The LAR indicated that general RMAs include:

- consideration of rescheduling maintenance to reduce risk,
- discussion of RICT in pre-job briefs,
- consideration of proactive return-to-service of other equipment, and

- efficient execution of maintenance.

The LAR discussed that configuration-specific RMAs are also developed in the CRMP to identify candidates to manage the risk associated with internal events, internal flooding, and fire events. These actions include:

- identification of important equipment or trains for protection,
- identification of important Operator Actions for briefings,
- identification of key fire initiators and fire zones for RMAs in accordance with the site Fire RMA process,
- identification of dominant initiating events and actions to minimize potential for initiators, and
- consideration of insights from PRA model cutsets, through comparison of importance.

Further, the LAR discussed that common cause RMA candidates include:

- performance of non-intrusive inspections on alternate trains,
- confidence runs performed for standby SSCs,
- increased monitoring for running components,
- expansion of monitoring for running components,
- deferring maintenance and testing activities that could generate an initiating event which would require operation of potentially affected SSCs,
- readiness of operators and maintenance to respond to additional failures, and
- shift briefs or standing orders which focus on initiating event response or loss of potentially affected SSCs.

The LAR explained that per fleet procedures if an emergent condition occurs for which a extent of condition cannot be assessed prior to entering into the RMAT or the extent of condition assessment cannot rule out the potential for CCF, then RMAs are expected to be implemented to mitigate CCF and its impact. The LAR also stated that these RMAs can include pre-identified RMAs as described above as well system specific RMAs. The NRC staff concludes the licensee's process for developing RMAs is in accordance with NEI 06-09-A because it uses configuration-specific risk insights and specifically considers the potential for CCFs in emergent conditions.

Based on the incorporation of NEI 06-09-A in the TS as discussed in LAR Attachment 1 and use of RMAs as discussed in LAR Enclosure 12, "Risk Management Actions" and the RAI supplemental information, and because the proposed changes are consistent with the guidance of RG 1.174, Revision 3; and RG 1.177, Revision 1, the NRC staff finds the 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. Thus, the RICT Program provides an acceptable methodology to assess and address risk-significant configurations. Further, reassessment of any plant configuration changes is also required 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 licensee's incorporation of NEI 06-09-A in the TS, as discussed LAR Attachment 1 and use of RMAs as discussed in LAR Enclosure 12, "Risk Management Action Examples," 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 LAR has demonstrated the technical acceptability and scope of its PRA models, and that the models can support implementation of the RICT Program for determining CTs. The LAR provided the appropriate key assumptions and sources of uncertainty. The risk metrics are consistent with the approved methodology of NEI 06-09-A and the acceptance guidance in RG 1.177 and RG 1.174. 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, "Monitoring Program," the SSCs in the scope of the RICT Program are also in the scope of the Maintenance Rule. 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. LAR Enclosure 11 states that cumulative risk will be calculated at least every two refueling cycles not to exceed 24 months, which is consistent with NEI 06-09-A. The RICT Program will convert 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 will be 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 the RICT program tracking feature as described in NEI 06-09-A, Section 3.3.3. The RICT program's risk-tracking feature is therefore acceptable and complies with this RMTS Program.

3.1.5.1 Key Principle 5 Conclusions

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 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 3, have been met and concludes that the proposed variations are acceptable.

3.2.1 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 in either unit at Prairie Island or the way that they are 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 limiting condition for operation 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.

3.2.2 Conditions Requiring Additional Technical Justification

Table 1, "Conditions Requiring Additional Technical Justification," of TSTF-505, Revision 2, contains a list of Required Actions that may be proposed for inclusion in the RICT Program. However, additional technical justification is required to explain why the Condition would not represent a loss of specified safety function as used in the RICT Program. Suggestions are provided in the table, but the suggestions may not be all encompassing for all plants. Licensees should provide sufficient information when adopting the listed Required Actions to justify that the condition does not represent a loss of specified safety function as used in the RICT Program.

The following excerpt from Table 1 presents the additional evaluations applicable to proposed changes to Containment Systems or Plant Systems TSs:

Table 1 of TSTF-505, Rev. 2
Conditions Requiring Additional Technical Justification

Specification	LCO Requirements and Condition	Suggested Information
3.6.2.C	LCO: [Two] containment air lock[s] shall be OPERABLE. Condition: One or more containment air locks inoperable for reasons other than an inoperable door or inoperable interlock mechanism.	Licensee must justify that an inoperable containment air lock is not a condition in which all required trains or subsystems of a TS required system are inoperable. An acceptable argument may be that a note in TS 3.6.2 requires the condition to be assessed in accordance with TS 3.6.1, Containment Integrity, and excessive leakage would require an immediate plant shutdown under that TS.
3.7.2.A	LCO: [Four] MSIVs shall be OPERABLE. Condition: One MSIV inoperable in MODE 1.	Licensee must justify that the condition would not prevent performance of the steam line break isolation function assumed in the accident analysis. An acceptable method may be a second MSIV per steam line, another design feature, or an alternate method of preventing blowdown of more than one steam generator.

<u>Note 1</u>

As discussed in Section 2.3 of the justification, some Conditions are applicable when an unspecified number of subsystems or instrument channels are inoperable, typically written as “One or more…” or “Two or more…”. These conditions currently apply when all subsystems or channels required to be operable to perform a function are inoperable, and application of a RICT in this situation is prohibited.

To address this, the following modification should be made to the CTs potentially applicable when all required subsystems or channels are inoperable.

72 hours {i.e., the existing Completion Time}

OR

---------------------NOTE--------------
Not applicable when
[all/two/four/both, etc.] required
[channels/subsystems/trains, etc.]
are inoperable.

In accordance with the Risk
Informed Completion Time
Program

The bracketed description will depend on the specification. This approach retains the existing requirements and limits the use of a RICT to conditions in

> which the function can still be performed. The licensee must justify that the required function can still be performed absent an additional failure when a RICT is applied. The TS Bases should be revised to describe the Note and the selection of the minimum number of channels needed to perform the function.

The following information was obtained from the LAR and USAR, related to the containment and plant systems TSs where the condition could involve a loss of function without additional specified conditions or limitations:

Condition 3.6.2.C, "One or more containment air locks inoperable for reasons other than Condition A or B," Required Action C.3 "Restore air lock to OPERABLE status"

As indicated in Table E1-1 of Enclosure 1 of the LAR, the containment air locks are explicitly modeled in the Prairie Island PRA. The PRA success criterion is one of two containment air lock doors closed with acceptable containment leakage per LCO 3.6.1. Specifically, the licensee identified that:

> [c]ompliance with the remaining portions of LCO Condition 3.6.2 ensure that there is a physical barrier (i.e., closed door) and an acceptable overall leakage from containment. Thus, the function is still maintained. Required Action C.1 of LCO Condition 3.6.2 requires the condition to be assessed in accordance with TS 3.6.1, "Containment Integrity" (i.e., "Initiate action to evaluate overall containment leakage rate per LCO 3.6.1" with a CT of Immediately). Therefore, the LCO meets the listed requirements for inclusion in the RICT Program.

Condition 3.7.2.A, "One MSIV inoperable in MODE 1," Required Action A.1 "Restore MSIV to OPERABLE status"

As indicated in Table E1-1 of Enclosure 1 of the LAR, the MSIVs are explicitly modeled in the Prairie Island PRA. The PRA success criteria depend on the accident scenario. The steam line break isolation function assumed in the accident analysis require the non-return check valves to isolate the affected SG and closure of the MSIVs are not modeled.

As required by Prairie Island TS 3.7.2, with one MSIV inoperable, action must be taken to restore to operable status within 8 hours.

As described in Section 14.5.5.2, "Expected Plant Response," of the Prairie Island USAR, Revision 35 (ADAMS Accession No. ML18166A204), each steam line has a fast-closing isolation valve with a downstream check valve. Section 14.5.5.2 of the USAR provides further information regarding an alternate method of preventing blowdown of both steam generators:

> These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, in the case of a break upstream of the isolation valve in one line, closure of either check valve in that line or the isolation valve in the other line will prevent blowdown of the other steam generator. In particular, the arrangement precludes blowdown of more than one steam generator inside the containment and thus prevents structural damage to the containment.

The NRC staff evaluated the information provided for these selected Conditions where a loss of function is not clearly precluded by the Condition statement. Including a RICT for the relevant Conditions is based on appropriate constraints to preclude application of a RICT when a loss of function condition exists.

For Condition 3.6.2.C, which applies when an air lock is inoperable for reasons other than an inoperable door or an inoperable interlock mechanism, the potential for containment leakage beyond allowable limits must be assessed to ensure no loss of containment function is associated with the air lock inoperability. The licensee's proposed change to the associated Action C.3 permits consideration of the RICT only when action to evaluate overall containment leakage rate per TS 3.6.1 has been immediately initiated and one air lock door is closed. These conditions provide reasonable assurance that any loss of function condition would be detected and preclude usage of the RICT. Therefore, the proposed change to Action 3.6.2.C.3 is acceptable.

For Condition 3.7.2.A, which applies when one MSIV is inoperable in Mode 1, verification of the failure of one MSIV has been evaluated if the main steam line break accident analysis is necessary to assure that accident assumptions would be satisfied under the specified condition to maintain design function. The accident analysis assumes the failure of one MSIV, which would allow the blowdown of one steam generator after the period assumed for closure of the operable MSIV. Therefore, there is no loss of function condition associated with the condition, and a RICT is appropriate.

3.2.2 Plant-Specific LCOs

In Attachment 4 of the LAR, the licensee identified and provided additional justification for several plant-specific LCOs and associated Actions, for which the LAR proposed applying the RICT Program, that are variations from TSTF-505, Revision 2, as discussed below:

Condition 3.6.5.C, "One or both containment cooling fan coil unit(s) (FCU) in one train inoperable," Required Action C.1 "Restore containment cooling FCU(s) to OPERABLE status and Condition 3.6.5.D, "One containment cooling FCU in each train inoperable," Required Action D.2 "Restore all FCUs to OPERABLE status."

Proposed TS 3.6.5 Condition C is based upon Condition C of the NUREG-1431 STS 3.6.6A, which is within the scope of TSTF-505, Revision 2. Proposed TS 3.6.5 Condition D is a plant specific Condition. Proposed TS Conditions C and D are unique in that they are based on combinations of individual FCUs inoperable within the trains of containment cooling, while STS 3.6.6A Condition C is in terms of trains of containment cooling inoperable. Proposed TS 3.6.5 Condition C is for one or both of the containment cooling FCU(s) in one train inoperable with the Required Action to restore the inoperable FCU(s) to OPERABLE status within 7 days. Condition D is for one containment cooling FCU inoperable in each train with the Required Actions to initiate action to isolate both inoperable FCUs immediately and restore all FCUs to OPERABLE status within 7 days. The 7-day CTs were developed taking into account the heat removal capabilities afforded by combinations of the CS System and Containment Cooling System and the low probability of a design basis accident occurring during this period.

Prairie Island has two trains of containment cooling with two FCUs per train. Each train of containment cooling has sufficient capacity to supply 100 percent of the Containment Cooling System design cooling requirements. Additionally, any two FCUs from opposite trains are

capable of providing the safety function, post-accident containment cooling, if cooling water flow to the inoperable FCUs is isolated.

As indicated in Table E1-1 of Enclosure 1 of the LAR, a hydraulic analysis has been performed to show that success or failure of FCUs does not impact which core damage sequences are classified as contributing to LERF. Therefore, the FCUs are not explicitly modeled in the Prairie Island PRA.

The Containment Cooling System is an Engineered Safety Feature (ESF) system. The ESF is designed to ensure that the heat removal capability required during the post-accident period can be attained. One train of containment cooling with one train of CS can provide 100 percent of the required peak cooling capacity during post-accident conditions.

Therefore, the NRC staff finds that TS 3.6.5 Conditions C and D are consistent with TSTF-505, Revision 2, and are, therefore, acceptable for inclusion in the RICT Program.

Condition 3.7.8.A, "No safeguards CL pumps operable for one train" Required Action A.1 "Restore one safeguards CL pump to OPERABLE status" and Condition 3.7.8.B, "One CL supply header inoperable" Required Action B.3 "Restore CL supply header to OPERABLE status."

Proposed TS 3.7.8 Condition A is a plant-specific Condition not in the NUREG-1431 STS or TSTF-505, Revision 2. Condition A is for the condition of no safeguards CL pumps OPERABLE for one train with action to restore one CL safeguards pump to OPERABLE status within 7 days. Either one of two diesel-driven CL pumps for the train may be restored to OPERABLE status, or the vertical motor-driven CL pump may be aligned to fulfill the safeguards function for the train that has no OPERABLE safeguards CL pump.

The diesel-driven cooling water pumps, vertical motor-driven pump and their associated equipment are located in the Class I portion of the cooling water screenhouse and separated by a concrete wall. A ring header which is shared by the units can be isolated automatically to provide two redundant independent sources of cooling water for all essential services. One-half of essential services for each Unit is supplied from each side of the isolable loop. Each side of the loop is designed to supply the needs for all essential services for both units. Thus, failure of one side of the loop still provides for the operation of all equipment required for the safe shutdown of both units.

The Prairie Island CL system is a shared system between units with a design basis to maintain cooling for the heat loads of one unit in MODE 3 and the second unit in long term post-accident condition. One CL train, in conjunction with the component cooling water system and a 100 percent capacity containment cooling system, has the capability to remove long term core decay heat following a design-basis LOCA as discussed in the Section 6 of the Prairie Island USAR. The 7-day CT is based on the low probability of loss of offsite power during the period; the low probability of a design-basis accident occurring during this time period; the safeguards cooling capabilities afforded by the remaining OPERABLE train; and the capability to route water from the non-safeguards pumps, if needed.

Note 3 of Condition A specifies that "no safeguards CL pumps OPERABLE for one train" may not exist for more than 7 days in any consecutive 30-day period. The LAR proposed to delete this note as risk will be adequately managed through both the application of the RICT Program,

as well as existing programs such as the Maintenance Rule and the monitoring of Mitigating Systems Performance Index.

Proposed TS 3.7.8 Condition B is based on STS 3.7.8 Condition A, which is in-scope of TSTF-505, Revision 2. Condition B is for the condition of one CL supply header inoperable with the Required Action B.3 to restore the CL supply header to OPERABLE status within 72 hours. The 72-hour CT is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a design-basis accident occurring during this time period.

As indicated in Table E1-1 of Enclosure 1 of the LAR, the CL System is explicitly modeled in the Prairie Island PRA. The PRA Success Criterion is as follows for Condition A:

- Two of five CL pumps (safeguards and non-safeguards) to support all normal or accident loads in the ring-header configuration with no demand reduction.

- One of three (as applicable) CL pumps (safeguards and non-safeguards) per operating CL train to support all normal or accident loads in the split-header configuration with no demand reduction.

- One of five CL pumps (safeguards and non-safeguards) to support the Unit 1 DG operation in the short-term with the ring-header configuration.

- One of five CL pumps (safeguards and non-safeguards) for both CL trains to support all accident loads in the long-term after reducing demand from normal loads.

- The PRA success criterion for Condition B is one of two supply headers for Condition B.

Therefore, the NRC staff finds that inclusion of TS 3.7.8 Conditions A and B in the RICT Program acceptable because there is no loss of function condition associated with the condition, and a RICT is appropriate.

<u>Plant-Specific LCOs Conclusion</u>

The NRC staff evaluated the information provided for these plant-specific LCOs and associated Actions to confirm that the condition does not represent a TS loss of function. Including a RICT for the relevant Conditions is based on appropriate constraints to preclude application of a RICT when a loss of function condition exists.

For Conditions 3.6.5.C and 3.6.5.D, which applies when one or both FCUs in one train is inoperable, or when one FCU in each containment cooling train is inoperable, the LAR stated that in this degraded condition the remaining operable containment spray and cooling trains provide iodine removal capabilities and are capable of providing at least 100 percent of the heat removal needs to maintain design function. Therefore, there is no loss of function condition associated with the condition, and a RICT is appropriate.

For Condition 3.7.8.A and 3.7.8.B, which applies to when no safeguards CL pumps are operable for one train, or when one CL supply header is inoperable, the LAR described the CL System as a shared system between units with a design basis to maintain cooling for the heat loads of one unit in MODE 3 and the second unit in long-term post-accident conditions. The LAR stated that with one CL train, in conjunction with the Component Cooling Water System and a 100 percent

capacity containment cooling system, has the capability to remove long term core decay heat following a design-basis LOCA as discussed in the Section 6 of the Prairie Island USAR. Therefore, there is no loss of function condition associated with the condition, and a RICT is appropriate.

3.3 Technical Specification Administrative Controls Section

The NRC staff reviewed the proposed addition of a new program, the RICT Program, to the Administrative Controls section of the Prairie Island TSs. 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.

Technical Specification 5.5.18 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 Operational Conditions (or 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 TSs contain requirements for the treatment of CCFs for emergent conditions in which the common cause evaluation is not complete. The requirements are to either (a) numerically account for the increased probability of CCF or (b) 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 defense-in-depth philosophy. The seven considerations supporting the evaluation of the impact of the change on defense-in-depth 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 defense-in-depth because the methods limit the exposure time or ensure the availability of alternate SSCs.

The TSs contain 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.

Technical Specification 5.5.18 was updated to reflect Revision 2 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 5.5.18 restates this limitation and condition from the NRC staff's SE in language that is appropriate for the Administrative Controls section of the Prairie Island 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 Prairie Island 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 Administrative Controls section of the TS is 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.

4.0 SUMMARY

4.1 NRC Staff Findings and Conclusions

The NRC staff finds that the proposed implementation at Prairie Island of the RICT Program for the identified scope of RMAs is consistent with the guidance of NEI 06-09-A. The licensee's methodology for assessing the risk impact of extended CTs, including the individual CT extension impacts in terms of ICDP and ILERP, and the overall program impact in terms of ΔCDF and ΔLERF, is accomplished using PRA models of sufficient scope and technical adequacy based on consistency with the guidance of RG 1.200, Revision 2 with completion of the implementation items. The RICT calculation uses the PRA model as translated into the CRMP model, and the licensee has an acceptable process in place to ensure the PRA model continues to use NRC accepted methods and is appropriately updated to reflect changes to the plant or operating experience. In addition, the NRC staff finds that the proposed implementation of the RICT Program addresses the RG 1.177 defense-in-depth philosophy and safety margins to ensure that they are adequately maintained and includes adequate administrative controls as well as performance monitoring programs.

4.2 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 3.

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), 50.46 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 5.5.18 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 3; 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 Prairie Island 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 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. 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 3, 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.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations on, the Minnesota State official was notified of the proposed issuance of the amendments on December 17, 2020. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on February 25, 2020 (85 FR 10733). Accordingly, the amendments meet 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 amendments.

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Circle, NRR
S. Dinsmore, NRR
B. Lee, NRR
D. Woodyatt, NRR
A. Russell, NRR
S. Basturescu, NRR
S. Wyman, NRR
K. Tetter, NRR

Date of Issuance: March 15, 2021

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 235 AND 223 RE: ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE (TSTF) TRAVELER TSTF-505, REVISION 2, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES – RITSTF INITIATIVE 4b" (EPID L-2019-LLA-0283) DATED MARCH 15, 2021

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NAME	RKuntz	SRohrer (JBurkhardt for)	RPascarelli	BWittick
DATE	12/17/2020	12/17/2020	10/29/2020	11/03/2020
OFFICE	NRR/DRA/APLC/BC	NRR/DEX/EICB/BC	NRR/DEX/EEEB/BC	NRR/DSS/STSB/BC
NAME	SRosenberg	MWaters	BTitus	VCusumano
DATE	10/26/2020	10/21/2020	12/08/2020	11/02/2020
OFFICE	NRR/DSS/SNSB/BC	OGC NLO	NRR/DORL/LPL3/BC	NRR/DORL/LPL3/PM
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